Viability Assessment of a Repository at Yucca Mountain

Preliminary Design Concept for the Repository and Waste Package

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Viability Assessment of a Repository at Yucca Mountain

Volume 2: Preliminary Design Concept for the Repository and Waste Package

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ACRONYMS

ASTM	American Society for Testing and Materials
CFR	*Code of Federal Regulations*
CRWMS	Civilian Radioactive Waste Management System
DOE	U.S. Department of Energy
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
LA	License Application
M&O	Management and Operating Contractor
NRC	Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management
TSPA	Total System Performance Assessment
VA	Viability Assessment
YMP	Yucca Mountain Site Characterization Project

Measurements

Btu	British thermal unit
cm	centimeter
DbA	decibel
Eh	redox potential
ft	foot
g	gram
hp	horsepower
in.	inch
kg	kilogram
km	kilometer
kPa	kilopascal
kV	kilovolt
kVA	kilovolt-ampere
kW	kilowatt
lin ft	linear feet
m	meter
mL	milliliter
mm	millimeter
MPa	megapascal
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
MVA	megavolt-ampere

nm	nanometer
pH	hydrogen-ion concentration notation
ppm	parts per million
ppmv	parts per million by volume
psi	pounds per square inch
rem	roentgen equivalent man (measure of absorbed dose)
wt	weight

OVERVIEW

The U.S. Congress directed the U.S. Department of Energy (DOE) to include in its Viability Assessment (VA) a discussion of the design for a Monitored Geologic Repository, including critical elements of repository surface, subsurface, and waste package designs. Taken together, the five volumes of the VA present a comprehensive and integrated description of the Yucca Mountain site, Monitored Geologic Repository and waste package designs, overall repository performance, plans to move forward from VA to License Application (LA), and associated costs.

Volume 2 contains the following:

- Descriptions of the critical elements of repository subsurface, waste package, and surface designs
- Discussions on the design process and design bases
- Descriptions of the surface and subsurface systems, including those addressing waste retrieval
- Description of the engineered barrier system, including waste package designs and design options, along with measures taken to extend the life of the waste package and protect the integrity of the spent nuclear fuel cladding
- Concepts for construction and operation
- Considerations regarding design flexibility
- Discussions of major design alternatives

Information presented in this volume is extracted from much of the design analyses, technical documents, and engineering drawings prepared under a quality assurance program approved by the Nuclear Regulatory Commission (NRC). These support documents provide in-depth technical treatment of the summary information presented in this document and can be accessed through DOE records systems. Appropriate references to such support documents are included throughout the text. The information presented in this volume reflects the design as of March of 1998. This design is referred to as the VA reference design.

It should be noted that the VA reference design described, discussed, and evaluated in the VA represents a "snapshot" of the design in its evolutionary process. The design will continue to progress as additional site data are gathered and additional performance assessments are conducted and evaluated. The integrated evaluations of these site data and performance assessments may result in further modifications to the VA reference design of the Monitored Geologic Repository and engineered barrier system.

CRITICAL ELEMENTS

Critical elements are those structures, systems, and components of the Monitored Geologic Repository that are relied upon to protect the public and/or workers from potential radiological risks during the operational lifetime of the repository or to protect public health and safety in the postclosure period. Critical elements for the subsurface, waste package, and surface designs are discussed in Section 1.2.

DESIGN PROCESS

The design process discussion in Section 2.1 presents the strategy for design development and discusses how this strategy has evolved over time. It includes a brief history of the evolution of the repository design since the release of the *Site Characterization Plan Overview, Yucca Mountain Site, Nevada Research and Development Area, Nevada* in December 1988 (DOE 1988a). The design has evolved during the last 10 years with the acquisition of a significant amount of site data, changes in the design philosophy regarding waste package size and configuration, excavation methods, and maturation of the process of predicting long-term performance of the overall system. Section 2.1.2 describes the design decision-making and documenting process as it relates to quality assurance requirements for design control. Section 2.1.3 discusses the Nuclear Quality Assurance program and its applicability to various aspects of the design.

The evaluation of Monitored Geologic Repository systems to ascertain their importance to radiological safety in the preclosure period (termed important to safety) and their potential to impact long-term postclosure performance (important to waste isolation) is underway. This evaluation is important not only because it determines the level of nuclear quality assurance applied to the various systems but also as a tool to prioritize ongoing design work. In general, those systems that are important to radiological safety or to waste isolation are given priority for resource allocation. The current status of system evaluations is presented in Section 2.2.

The current design effort is focused appropriately on those systems considered important to a radiologically safe operation and the successful long-term performance of the engineered barrier system. Systems that are not considered important to radiological safety or waste isolation, such as certain utility systems, administrative, and non-safety construction functions, are treated broadly in this volume. Non-radiological safety considerations such as silica dust control and other occupational safety considerations are considered equally important but are not addressed in this volume of the VA.

The process of design prioritization, called binning, is described in Section 2.3. Binning is the placement of the various systems in one of three possible bins according to the system's radiological or waste isolation importance and regulatory precedent. Bin 3 systems have nuclear safety implications and no regulatory precedent. In this context, a lack of regulatory precedent means that NRC has no previous history in licensing this type of system. Bin 2 systems have nuclear safety implications and licensing precedent. Accordingly, Bin 3 systems have highest priority, followed by Bin 2 systems. Bin 1 systems have no radiological significance and, therefore, the lowest priority at this point in the design evolution process. These non-radiological aspects of design will be developed prior to construction authorization.

DESIGN BASES

Section 3 contains discussions of the following bases for repository design:

- Design requirements
- Primary design assumptions that are in the process of being verified
- Required postclosure functions of the systems
- Major preclosure goals and objectives

The regulatory requirements for the repository system are captured primarily in the *Repository Design Requirements Document* (YMP/CM 0023, REV. 00, ICN 01) (YMP 1994b) and the *Engineered Barrier Design Requirements Document* (YMP/CM 0024, REV. 00, ICN 01) (YMP 1994a). *The Nuclear Waste Policy Act of 1982*, as amended, and 10 *Code of Federal Regulations* (CFR) 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, are the primary sources for these requirements. Section 3.1 addresses design requirements for the repository system.

The current phase of repository design includes certain assumptions that will be the subject of ongoing and future analytical and site characterization work. The VA reference design is predicated on these assumptions regarding the likely outcome of this analytical work and characterization. These controlled design assumptions and the process for documenting, ensuring common usage of assumptions, and closing out these assumptions are discussed in Section 3.2.

Section 3.3 addresses functions that are allocated to various systems for long-term performance. The allocation of postclosure performance functions to a system means that a certain amount of reliance is placed on that system to perform as expected. The concept of multiple barriers having different failure modes (defense in depth) also is discussed. The concept of long-term performance is defined and discussed in Volume 3 of the VA.

Section 3.4 discusses preclosure radiological goals and objectives. The preclosure period includes the time people will routinely work in and around the repository. The radiological health and safety of the workers, as well as those individuals inhabiting the surrounding areas, is addressed in this section. Regulations applicable to the design also are discussed.

REPOSITORY DESIGN

The design of the Monitored Geologic Repository surface facilities and subsurface layout is discussed in Section 4. Relevant figures are included. Equipment and procedures for waste handling already approved by NRC and in use at other licensed nuclear facilities have been incorporated into the design and operating concepts to the maximum extent practicable. A Repository Design Consulting Board comprised of industry experts was assembled to review and comment on the design. The Board provided guidance in the areas of waste receipt, transfer, handling, and packaging; underground excavation methods; waste package design; and waste package materials testing. The waste package design and materials testing program are described in Section 5.

Major emphasis is intentionally placed on describing and discussing the underground systems and the engineered barrier system. Less description and discussion are provided for surface systems that will be similar to those already in use at other nuclear facilities licensed by NRC.

Radiologically significant repository surface facilities and their functions and operations are described in Section 4.1. The Waste Handling Building is the largest and most complex structure. It houses all facilities and equipment required to remove spent nuclear fuel assemblies and other high-level radioactive waste forms from their transportation casks; handle the casks and wastes within the facility; place the wastes into disposal containers; close and inspect the containers; and load the waste packages (loaded, sealed, and tested disposal containers) onto the conveyance that transports the waste packages underground for emplacement.

Other waste-related facilities also are discussed:

- Carrier Preparation Building
  - Processes transportation casks before their entrance into the Waste Handling Building
  - Readies empty casks and associated equipment for shipment back to the waste generators
- Waste Treatment Building
  - Treats site-generated low-level radioactive wastes

Some support-related facilities, called balance-of-plant, also are discussed briefly.

Section 4.2 contains descriptions of the repository subsurface design, including the total excavation requirements, and sequence of construction.

The repository subsurface layout, discussed in Section 4.2.1, consists of main drifts and emplacement drifts as discussed in Section 4.2.1.4. The repository host horizon is located above the water table in the dry, unsaturated zone, consisting of volcanic tuff, to take advantage of the features of the natural barrier. Main drifts provide travelways for equipment, personnel, ventilating air, and waste packages. Emplacement drifts are the tunnels in which the waste packages will be placed. Subsurface access is provided by two gently sloping ramps and two vertical shafts. Waste package transport into the subsurface facility is via the north ramp. No waste is moved into the subsurface facility via the vertical shafts.

The subsurface layout will be excavated using tunnel boring machines, with less than 5 percent by other mechanical means. No drill-and-blast excavation is planned currently. The existing 7.9-km (4.9-mile) tunnel that was constructed to assist in site characterization activities forms a part of the system of main drifts.

Transportation in the subsurface is provided by a rail system of conventional design. Power for subsurface vehicles is provided by a direct-current electric trolley system. During the period when emplacement and development operations are concurrent, the repository will be partitioned into two independent areas. One area encompasses waste emplacement operations while the other supports construction of waste emplacement drifts. Under normal conditions, no vehicular or personnel traffic will pass from one area to the other.

The ground control system is discussed in Section 4.2.2. This system, installed during the excavation of the facility, provides the means to ensure stability of the subsurface openings during the preclosure period. A robust lining system of precast concrete segments was selected as the primary ground support system for the emplacement drifts. Steel sets with steel lagging will be employed in about 10 percent of the emplacement drifts that are to be geologically mapped. Either system will require little or no maintenance over the preclosure period.

The waste transport and emplacement system, discussed in Section 4.2.3, moves the loaded waste packages to the subsurface and places the waste packages in the emplacement drifts.

The subsurface ventilation system, discussed in Section 4.2.4, consists of two separate and independent fan systems and flow networks separated by moveable air locks. One system provides air to the development operations area while the second system ventilates the waste emplacement operations area. Development of new emplacement areas and emplacement of waste in previously prepared areas take place simultaneously over a period of approximately 20 years. Air pressure in the development side is always higher than the pressure in the emplacement side. In the unlikely event that radioactive particulates are released into the subsurface airstream on the emplacement side, the pressure differential will prevent the spread of these particles to the development operations area. The emplacement-side ventilation system includes a high-efficiency particulate air filter system to trap and retain any radioactive particulates that may be released. The subsurface repository monitoring and control system, including certain performance confirmation functions, is discussed in Section 4.2.5.

Repository closure and decommissioning is discussed in Section 4.3. Subsurface closure will begin with the removal of nonpermanent items from the subsurface that will not be needed during the closure process. Backfill material, possibly sand or a portion of the rock initially removed during excavation of the underground facility, will be placed in the main drifts. The VA reference design does not include placement of backfill in the emplacement drifts. Use of backfill in the emplacement drifts is an option and is discussed in Section 5.3.1.

Seals will be installed at various locations in the ramps and shafts, with backfill material placed in the spaces between the seals. Plugs will be installed at the surface entrance to the ramps and shafts. The plugs and seals are designed to inhibit future human intrusion into the repository and prevent the ramps and shafts from providing preferential pathways for water to enter into the repository host horizon or for radionuclides to escape to the biosphere.

Closure of the surface facilities will start with decontamination of all radiological areas. Resulting low-level radioactive wastes will be compacted, packaged, and stored temporarily onsite until transported for disposal offsite. Structures will be emptied of equipment and dismantled. Debris will be disposed of in approved offsite facilities. Foundations and slabs will be removed and the surface returned to its approximate original condition. Native plants will be cultivated to complete the restoration.

Monuments are planned for various locations around the site. The monuments will serve to warn future inhabitants of the presence and nature of the emplaced wastes. Provisions may be added for postclosure monitoring.

The performance confirmation program, discussed in Section 4.4, includes elements of site testing, repository testing, repository subsurface support facilities construction, and waste package and materials testing.

ENGINEERED BARRIER SYSTEM DESIGN

The engineered barrier system, discussed in Section 5, is composed of the waste forms, the surrounding waste package, and other man-made items in the underground facility. The non-waste package components include the steel waste package supports; the piers upon which the supports rest; the tunnel bottom, or invert; and the linings installed in the emplacement drifts. Backfill, if used in the emplacement drift, would also be considered a part of the engineered barrier system (see Section 5.3.1). The designs for the waste package and remainder of the engineered barrier system were selected to complement the performance features of the natural barrier.

Section 5.1 addresses the components that make up the waste package, and the design aspects of these components. The role that spent nuclear fuel cladding plays as a barrier to radionuclide dissolution and the measures taken to protect the integrity of the cladding also are discussed. There are several waste package configurations in the VA reference design because of the variety in size, thermal output, origin, initial enrichment, and burnup characteristics of waste forms that must be accommodated. Commonality among waste package designs has been achieved to the maximum extent practicable. The waste package is the primary component of the engineered barrier system relied upon to contain radionuclides.

Performance assessments in the *U.S. Department of Energy (DOE) and U.S. Nuclear Regulatory Commission (NRC) Total System Performance Assessment (TSPA) Technical Exchange Summary Report* (DOE 1995b) have demonstrated that the amount of water contacting the waste packages is the most important determinant in assessing the ability of the site to contain and isolate high-level radioactive waste. Therefore, the emplacement drifts are located above the water table in the unsaturated zone.

The waste packages are composed of two concentric containment barriers. The outer barrier is composed of 10 cm (4 in.) of carbon steel (A516) while the inner barrier is made up of 2 cm (0.8 in.) of a high-nickel alloy American Society for Testing and Materials (ASTM) B 575 N06022 (also referred to as Alloy 22). The dual-metal-barrier design provides protection against two different and independent container degradation modes. The outer barrier is fabricated with a corrosion allowance material that will corrode slowly over time with little pitting or crevice corrosion. While the corrosion allowance material is intact and slowly corroding, it will protect the inner barrier material from exposure to oxygen and moisture. The dual-barrier design with differing degradation modes should remain intact for thousands of years. The use of a dual-barrier design represents the application of the defense in depth design philosophy. Use of dual barriers with different and independent failure modes supports the waste containment strategy by protecting against common mode failure that otherwise could occur with a single-material barrier.

The internal parts of the waste package depend on the type and characteristics of the waste placed in the package. Most of the waste packages will contain spent nuclear fuel assemblies with Zircaloy cladding from commercial power plants. There are two basic size groups of assemblies. The larger assemblies are from pressurized-water reactors and the smaller assemblies are from boiling-water reactors. While the pressurized-water reactor assemblies are larger in cross-section, the majority of both types are of approximately the same length. The internal package configuration for the commercial spent nuclear fuel assemblies consists of a metallic basket that holds the individual assemblies in place. The makeup of the basket can be changed to compensate for the nuclear criticality potential of the assemblies. Most assemblies need no criticality control and will have simple carbon-steel baskets. Some assemblies will need boron, which absorbs neutrons, in the steel of the baskets to prevent criticality.

An inner barrier to the dissolution and release of radionuclides is the metal cladding surrounding the ceramic-pellet waste form (see Section 5.1.1). Measures taken to protect the cladding are discussed in Section 5.1.3.2. The contribution of cladding to repository performance is discussed in Section 5.5.2 of Volume 3.

In addition to receiving commercial spent nuclear fuel, the repository will also accept vitrified high-level radioactive waste in stainless steel canisters. Vitrified waste is a glass waste form containing fission products concentrated from defense-related activities. Five canisters will be loaded into each waste package for disposal. A smaller diameter canister containing DOE-owned spent nuclear fuel from research reactor operations will be placed in the center of some of these waste packages. This concept is referred to as co-disposal and such canisters will be suitable for direct insertion into waste packages.

The waste packages will be placed on supports whose composition of carbon steel is identical to the composition of the outer barrier of the waste package. Carbon steel supports were chosen so as to alleviate galvanic reactions between the supports and the outer walls of the waste package in order to preclude rapid corrosion of the outer wall, thus promoting long waste package lifetimes. The pier is a steel box, containing concrete, with two vertical holes that accommodate the legs of the steel supports. The inverts, which support the piers, are composed of precast concrete. The invert serves as the floor of the emplacement drift and supports all operating equipment as well as the weight of the waste packages.

Evaluations to support the waste package design are discussed in Section 5.1.3. A significant, ongoing program of testing potential waste package materials is described, along with the testing program for degradation of the waste forms. Material performance modeling also is addressed.

Several options that may enhance the performance of the VA reference design engineered barrier system are being evaluated. These options are discussed in Section 5.3. Alternative waste package designs are discussed in Section 8.

CONCEPTS FOR CONSTRUCTION, OPERATION, MONITORING, AND CLOSURE

Section 6 contains descriptions of the construction, operation, monitoring, and closure of the facilities. Operations descriptions are provided, beginning with arrival of the waste in its transportation packaging, through receipt, preparation, and unloading of the wastes. Transfer of wastes, placement into disposal containers, and closure and testing of the containers are discussed.

Major subsurface operations are described, including the following:

- Construction of the initial portions of the subsurface repository prior to the start of waste emplacement operations
- Simultaneous operations of ongoing drift excavation and waste emplacement
- Waste package transport and emplacement
- Monitoring operations

The operation of the subsurface waste emplacement system is summarized in Section 6.2.2. Additional details on equipment and systems used are provided in Section 4.2.3.

DESIGN FLEXIBILITY CONSIDERATIONS

Complex systems are designed with a degree of flexibility so they can be operated over a wide range of conditions and accommodate future uncertainties. Section 7 addresses some of the major flexibility measures incorporated into the repository design.

Areas of flexibility include the use of multiple handling lines for commercial spent nuclear fuel and canistered waste forms to prevent a single breakdown of equipment from halting operations. Also, the surface facility will provide the capability to engage, open, and manipulate a wide variety of existing and future cask designs.

In the subsurface, flexibility will enable unexpected geologic conditions encountered during development of new emplacement areas to be addressed. Conducting emplacement operations simultaneously in multiple drifts will prevent a single equipment failure from shutting down operations and facilitate the emplacement of waste packages with a broad range of waste characteristics.

MAJOR DESIGN ALTERNATIVES

Sections 1 through 7 describe the VA reference design, as well as several options to the engineered barrier system design under evaluation. Section 8 discusses several major design alternatives that are significantly different from the VA reference design, in accordance with 10 CFR 60. These alternatives are under evaluation to determine their long-term performance and ascertain their costs. Evaluations of design options and major alternative designs will be completed prior to development of any LA. The work remaining to complete the evaluations and finalize the LA design is discussed in Section 3.2 of Volume 4.

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1. INTRODUCTION

1.1 SCOPE AND OBJECTIVE

This volume describes the major design features of the Monitored Geologic Repository. This document is not intended to provide an exhaustive, detailed description of the repository design. Rather, this document summarizes the major systems and primary elements of the design that are radiologically significant, and references the specific technical documents and design analyses wherein the details can be found.

Not all portions of the design are at the same level of completeness. Highest priority has been given to assigning resources to advance the design of the Monitored Geologic Repository features that are important to radiological safety and/or waste isolation and for which there is no NRC licensing precedent. Those features that are important to radiological safety and/or waste isolation, but for which there is an NRC precedent, receive second priority. Systems and features that have no impact on radiological safety or waste isolation receive the lowest priority. This prioritization process, referred to as binning, is discussed in more detail in Section 2.3.

Not every subject discussed in this volume is given equal treatment with regard to the level of detail provided. For example, less detail is provided for the surface facility design than for the subsurface and waste package designs. This different level of detail is intentional. Greater detail is provided for those functions, structures, systems, and components that play key roles with regard to protecting radiological health and safety and that are not common to existing nuclear facilities already licensed by NRC. A number of radiological subjects are not addressed in the VA, (e.g., environmental qualification of equipment). Environmental qualification of equipment and other radiological safety considerations will be addressed in the LA. Non-radiological safety considerations such as silica dust control and other occupational safety considerations are considered equally important but are not addressed in this volume of the VA (see Volume 1, Section 2.2.1.2, subsection on Health Related Mineral Issues).

1.1.1 Scope

Volume 2 describes the surface waste receiving and handling systems, subsurface layout and underground waste package handling and emplacement systems, and the disposal container (waste package) designs as of March 1998. This design is referred to as the VA reference design. The repository and disposal container designs are in the preliminary design stage, which is the second of three design stages: conceptual, preliminary, and final (also referred to as procurement and construction). The design will continue to evolve as issues are resolved, additional site data are acquired, performance assessments are reiterated and reevaluated, and important design bases are refined. The description presented in this volume represents a comprehensive and integrated picture of the overall VA reference design from a radiological safety perspective.

This volume also describes how the primary surface and subsurface facilities are intended to operate and identifies interfaces between the two facilities. Construction and operation of the subsurface facilities are closely interrelated and will occur simultaneously over an extended period. Therefore, the construction sequencing for the subsurface repository also is described. The loading, movement, and emplacement of waste packages are discussed within the context of the surface and subsurface structures, systems and components important to radiological safety and/or waste containment.

Important design bases for the Monitored Geologic Repository also are discussed in this volume. These design bases include regulatory requirements, basic assumptions, and waste characteristics. NRC key technical issues are addressed in Sections 1.2.1, 1.2.3, 2.2, 5.1, and 5.3.

Sections 1 through 7 focus on the VA reference design for the major surface, subsurface, and engineered barrier systems. Section 8 discusses major design alternatives being considered. These alternatives represent major departures from the basic concepts of the VA reference design. The intent of this discussion is to show what major alternatives exist, as well as to identify the potential benefits and limitations of each.

1.1.2 Objective

The objective of Volume 2 is to present a high-level summary of the radiological aspects of the Monitored Geologic Repository design based on the extensive work performed to date. The major design bases that have guided the design and the measures taken to extend the life of the waste package and protect the integrity of the spent nuclear fuel cladding are major contributors to long-term repository performance.

Volume 2 is intended to be a complement to the site description in Volume 1, the discussion of the Monitored Geologic Repository's long-term performance presented in Volume 3, the LA planning process described in Volume 4, and the cost information provided in Volume 5.

1.2 CRITICAL ELEMENTS

A systems engineering approach was used to develop the reference designs for the repository and waste packages. Technical baseline requirements were identified and functional requirements were developed and allocated to system architecture. Descriptions were prepared for each system which, among other things, identified interfaces among the various systems. These descriptions were included in the *Q-List* (DOE 1998a). Elements of the system designs were then input into performance assessments, as appropriate. The results of those assessments, which included engineering designs and the features of the natural system combined as inputs, were then evaluated again, the designs modified, and the performance assessments reiterated, as appropriate, to ensure robust designs. Thus, the approach taken in developing the VA reference designs reflects an integrated systems engineering approach. The design elements were then provided to cost engineers to estimate overall costs for the systems.

This section contains a summary of the critical elements of the repository subsurface, waste package, and surface designs. Critical elements are those structures, systems, and components of the Monitored Geologic Repository that play important roles in either preclosure safety or postclosure performance. That is, such structures, systems, and components are relied upon to protect the public and/or workers from potential radiological risks during the operational lifetime of the facility or to protect public health and safety in the postclosure period. Structures, systems, and components that are expected to have a radiological safety or waste isolation function, or other regulatory interest, without regulatory precedent have been categorized as Bin 3. Structures, systems, and components that are expected to have a radiological safety or waste isolation function, or other regulatory interest, with regulatory precedent have been categorized as Bin 2. There is a link between Critical Elements and Bin 2 and 3 categorized structures, systems, and components. A detailed description of binning and how binning is used to prioritize design activities is provided in Section 2.3, Prioritization of Design Activities.

The following three sections address critical elements and functions for the subsurface, waste package, and surface designs, respectively. As appropriate, the structures, systems, and components are discussed as they relate to the repository safety strategy and the Principal Factors of Expected Repository Performance discussed in Volume 4, Section 2. The repository safety strategy and the principal factors served as drivers in developing the VA reference design and provide a focus for future work leading to the preparation of a license application. The design aspects of that work are described in Sections 2 and 3 of Volume 4.

1.2.1 Repository Subsurface

There are 17 major systems that make up the subsurface repository architecture, of which 9 are expected to have a radiological safety or waste isolation function, or other regulatory interest, without regulatory precedent. The information in this section addresses aspects of the NRC Key Technical Issues of Structural Deformation and Seismicity (NRC 1997a), Evolution of the Near-Field Environment (NRC 1997b), Container Life and Source Term (NRC 1998a), Repository Design and Thermal-Mechanical Effects (NRC 1997c), and Radionuclide Transport (Sagar 1997). Detailed descriptions of subsurface systems functions and designs are presented in Section 4.2. As indicated in *Mined Geologic Disposal System Prioritization of Structures, Systems, and Components (SSCs),* nine of these were placed in Bin 3 (CRWMS M&O 1997m).

Subsurface System	Important To Preclosure Safety	Important to Postclosure Performance
Subsurface Facility System	X	X
Engineered Barrier System		X
Performance Confirmation System		X
Subsurface Closure and Sealing System		X
Ground Control System	X	
Subsurface Ventilation System	X	
Waste Emplacement System	X	
Waste Retrieval System	X	
Subsurface Central Control System	X	

1.2.1.1 Systems Contributing to Postclosure Performance

Subsurface Facility System. The subsurface facility system encompasses the location, size, and spacing of the underground openings, including access ramps and development, emplacement, and ventilation drifts that facilitate underground development, emplacement, and retrieval operations. Location of the subsurface facility system in the unsaturated zone helps to limit the water available to come in contact with the waste packages. The system protects the engineered barrier system and supports long waste package lifetimes and thus long-term waste isolation. The location of the subsurface facility system in the unsaturated zone helps to limit the water available to come in contact with the waste package. Thermal heating of the waste packages and the surrounding rock will create a low-humidity environment in the emplacement drifts. The emplacement drifts are located to avoid disruption by fault movement by maintaining a suitable stand-off distance from Type I faults. The subsurface facility system houses other subsurface systems such as ventilation, utility, safety, monitoring, and underground transport of waste packages and interfaces with the engineered barrier system, the natural barrier for the geologic setting, and the performance confirmation system. The repository subsurface facilities are discussed further in Section 4.2. Major alternatives to the subsurface facility system design under consideration are discussed in Section 8.

Engineered Barrier System. The engineered barrier system consists collectively of the waste packages, including spent nuclear fuel cladding; waste package support hardware; and performance enhancing barriers and is located entirely within the underground facility. The engineered barrier system benefits from a low-humidity environment during the first several thousand years achieved by drift layout and orientation and the thermal loading of the drift through waste package spacing within and among the emplacement drifts. The engineered barrier system supports long waste package lifetimes through the use of a concentric, dual-barrier, dual-materials design with different failure modes for the waste package. Thus, this system protects and slows the degradation of the waste form and the subsequent release and transport of radionuclides to the natural barrier for thousands of years, allowing for a reduction in radiation sources through radioactive decay. This, in turn, contributes to a reduced peak dose rate. Thus, the waste package component of the engineered barrier system works in concert with the natural geologic setting to delay, dilute, and diffuse radionuclide releases to the biosphere. If necessary, performance of the waste packages may be enhanced by the addition of drip shields, backfill, and ceramic coatings to prevent dripping water from contacting waste packages and to further delay corrosion of the dual-barrier waste packages. The engineered barrier system interfaces with the natural barrier, the subsurface facility system, and the ground control system. The engineered barrier system is discussed further in Section 5. Design options being evaluated, which may enhance the performance of the engineered barrier system, are discussed in Section 5.3. Major alternatives to the engineered barrier system design under consideration are discussed in Section 8. Options and/or alternatives will be used to modify the design, as appropriate, based on the work to be completed between VA and submittal of an LA (see Volume 4, Section 3.2).

Performance Confirmation System. The performance confirmation system acquires pertinent data associated with verifying the near-term and the long-term performance of the repository. It will verify that the natural and engineered systems and components are functioning as intended. The system will operate from the site characterization phase to the point of closure of the subsurface repository. This system operates in the subsurface repository acquiring and transmitting data to the surface. The performance confirmation system also operates on the surface acquiring data, conducting field and laboratory experiments, and analyzing data acquired from surface and subsurface operations. The system will provide data on the thermal, hydrologic, mechanical, and chemical changes that the repository will experience during the preclosure period. The performance confirmation system evaluates the accuracy and adequacy of the information used to determine that the repository will perform as expected in the long term (see Section 4.4). Portions of the performance confirmation system are discussed in Sections 4.2.5.3 and 5.1.4.

Subsurface Closure and Sealing System. The subsurface closure and sealing system provides closure barriers and seals for the underground openings, including surface and subsurface boreholes. Closure and sealing of such openings serve to minimize the amount of water capable of entering the repository host horizon through such openings, thus extending waste package lifetimes and delaying onset of waste form degradation. The system includes equipment and seal materials. The subsurface closure and sealing system helps to limit the amount of gases released from the repository into the biosphere. The system also serves to discourage human intrusion in the postclosure period.

The seal materials meet requirements for degradation, permeability, environmental conditions, chemical content, and pH levels consistent with ensuring acceptable performance assessment so that the seals will not serve as preferential pathways that could compromise overall repository performance. The subsurface closure and sealing system is discussed in Section 4.3.

1.2.1.2 Systems Contributing to Worker/ Public Health and Safety

Ground Control System. The ground control system limits the size, frequency, and extent of rockfall occurring in the underground openings, including access, ventilation, and emplacement drifts during the operational and monitoring periods for the Monitored Geologic Repository. Thus, the system provides protection for workers and the waste packages. The system also controls the rate of closure of openings caused by geomechanical, thermomechanical, and excavation effects, thus helping to extend the performance monitoring period and preserve the option to retrieve the waste should such retrieval prove necessary or desirable.

The system includes combinations of precast concrete lining, cast-in-place concrete lining, steel sets and lagging, and rockbolt and mesh supports. The ground control system provides protection to workers and equipment during the operations period. The system is designed to provide robust support with minimal maintenance requirements during the preclosure period, thus supporting performance confirmation activities. The use of steel sets and steel lagging for ground support in certain emplacement drifts facilitates the collection of additional geologic information. The ground control system interfaces with the subsurface facility system, the engineered barrier system, the waste emplacement system, the subsurface ventilation system, and the waste retrieval system. The ground control system is discussed further in Section 4.2.2. The performance of the system is discussed in Sections 2.4 of Volume 3. Major alternatives to the ground control system design under consideration are discussed in Section 8 of this volume.

Subsurface Ventilation System. The subsurface ventilation system is segmented into two, separate and independent subsystems. Moveable air locks are used to segment the emplacement-drift airflow from the development-drift airflow. The use of moveable air locks allows the boundary separating the two to be easily moved, as needed. The development side subsurface ventilation subsystem performs no nuclear safety function. It provides air and ventilation for excavation and development of emplacement drifts prior to waste package emplacement. The emplacement side subsurface ventilation subsystem provides ventilation for emplacement drifts that have been loaded with waste packages. The subsystem uses redundant fans (located on the surface) to maintain an air flow path that draws air from areas of less potential contamination to areas of greater potential contamination while monitoring the air for radioactive particulate matter. Upon detection of such particulates, the air flow is directed through redundant high-efficiency particulate air filters designed to remove and isolate such contaminants from the air prior to its release to the biosphere. Movement of air, heated by waste packages, through the emplacement drifts also helps to remove moisture contained in drifts and the rock mass and thus lowers humidity. This may contribute to the delayed onset of corrosion of the outer layer of the waste packages and subsequent degradation of the waste form. Prior to and during any waste retrieval operations, the subsurface ventilation system would be used to reduce the temperature inside the drifts. The subsurface ventilation system is discussed further in Section 4.2.4. Major alternatives to the subsurface ventilation system design under consideration are discussed in Section 8.

Waste Emplacement System. The waste emplacement system transports loaded and sealed waste packages from the North Portal of the Waste Handling Building to the subsurface emplacement drifts. System components include the following: electrically powered transport locomotives capable of manual and remote operations; a radiologically shielded 168-ton waste package transporter; and an electrically powered, remotely operated emplacement gantry which takes the waste package from the waste package transporter and places the waste package within the drift. Shielding on the waste package transporter and the capability for remote operations help to control worker radiation exposures. The waste emplacement system interfaces with virtually all other major subsurface systems. Components of the system are discussed in Section 4.2.3. Activities associated with the waste emplacement system are discussed in Section 6.2. Major alternatives affecting the waste emplacement system design are discussed in Section 8.

Waste Retrieval System. The waste retrieval system removes some or all of the waste packages from the emplacement drifts and transports the waste packages to the surface should such retrievable prove necessary or desirable. The system prepares the emplacement drift for retrieval and acquires the necessary waste package information and prepares the waste packages for transportation and transports them to the surface. The waste retrieval system restores the emplacement drift condition, as necessary, and inspects and services radiological systems prior to removal of waste packages. By maintaining the capability to remove all or any portion of the waste packages, the waste retrieval system is a potential contributor to the protection of the public health and safety during the operations and monitoring periods. The system interfaces with the subsurface facility system, the subsurface ventilation system, the ground control system, and the surface waste handling facilities. The waste retrieval system is discussed in Section 4.2.7.

Subsurface Central Control System. The subsurface central control system provides the capability to operate remotely both nuclear- and non-nuclear-related systems. The operation of this system is critical to the safe conduct of repository operations. This system is discussed in Section 4.2.5.

1.2.2 Waste Package

The engineered barrier system is one of the critical elements of the repository. Performance assessments have demonstrated that the amount of water contacting the waste packages is the most important determinant in assessing the ability of the site to contain high-level radioactive waste.

The primary component of the engineered barrier system is the waste package. The waste package has been designed to take advantage of locating the repository host horizon above the water table in the unsaturated zone, which consists of relatively dry rock. As long as the waste packages remain intact, the waste will be contained completely within the packages and prevented from contact with the host rock, air, or groundwater. A number of waste package designs have been developed to accommodate the variability in waste forms, sizes, and nuclear characteristics. Each has been evaluated and designated Bin 3 because of the importance of waste package designs to long-term repository performance and the lack of licensing precedent. The concept of binning is developed more fully in Section 2.3. Waste package designs and materials selection are discussed in Section 5. The waste packages are significant cost contributors.

Waste Package System	Important To Preclosure Safety	Important to Postclosure Performance
Waste Package and Supports	X	X

Over the repository lifetime, the waste package containment barriers will perform various functions that will change with time. The engineered barrier system begins performing its function of containing the waste during the 100-year-or-longer preclosure phase, supplemented by engineering and institutional controls required for handling, emplacing, and, if necessary, retrieving the waste. During the first several thousand years following repository closure, the containment barriers alone will either contain radionuclides completely or impede the transport by water of any radionuclides from waste packages that may have breached. However, even breached containment barriers are expected to inhibit the movement of water into and out of the waste packages, thereby limiting the transport of radionuclides to the biosphere. Detailed descriptions of the waste package functions and designs are presented in Sections 3.3 and 5.1, respectively. The waste containment function is divided into the following subfunctions:

- Confine the waste
- Limit the radionuclide release to the natural barrier
- Limit the radionuclide release to the accessible environment
- Limit the natural and induced environmental effects

1.2.3 Surface

This section provides descriptions of the significant waste handling facilities, systems, and equipment located in the radiologically controlled area. The radiologically controlled area is located within the Monitored Geologic Repository site controlled area, and refers to that surface area protected by fences and security systems where the radiological waste shipments are handled and the site-generated low-level radioactive water is processed.

The surface waste handling facilities perform the primary functions required to safely receive and prepare the waste shipped to the site, load the waste into disposal containers (waste packages), prepare them for underground transport and emplacement, and package the site-generated low-level radioactive waste for disposal offsite.

The production requirements for the waste handling operations are to receive the waste shipments at a rate defined by the shipping schedule, and to load the waste in disposal containers at a rate defined by the emplacement schedule. During the waste preparation activities, the shipping casks containing the waste will be handled inside and outside the facilities. The waste forms inside the casks, including spent nuclear fuel assemblies and canisters of waste, will be removed from the casks and loaded into disposal containers in the Waste Handling Building. The canisters and disposal containers holding waste are radioactive beyond the limits of direct contact by operating personnel. The facilities associated with handling the waste will be designed to shield the operating personnel from radiation and to prevent the release of radioactive effluents to the environment and the public. The facilities and systems associated with these operations will be designed to perform these operations during normal and off-normal events. Off-normal events will include the design basis for the repository and include loss of power, earthquake, flood, fire, and extreme weather conditions. In summary, the repository surface facilities must perform the following critical functions during the waste emplacement operations phase:

- Receive and prepare the waste for loading into disposal containers
- Prepare empty transportation casks for return to waste generators
- Load the waste into disposal containers, seal and test the containers
- Maintain the waste receipt and emplacement schedules
- Protect the operating personnel, environment, and public from radiation and hazardous substances
- Collect and package site-generated low-level radioactive waste and hazardous wastes for subsequent disposal offsite
- Maintain safe conditions and radiation confinement during normal, off-normal, and design basis events

There are 22 major systems that make up the surface repository architecture, of which four are expected to have a radiological safety or other regulatory interest, without regulatory precedent.

Surface System	Important To Preclosure Safety
Canister Transfer System	X
Assembly Transfer System	X
Waste Package Disposal Container	X
Disposal Container Handling System	X

None of the surface facilities' systems play a role in long-term performance of the repository. Detailed descriptions of surface systems functions and designs are presented in 4.1.

Canister Transfer System. The canister transfer system will receive shipping casks containing large and small disposable canisters, transfer the canisters from the casks into disposal containers, and prepare the empty casks for reshipment. Two identical remotely operated and shielded canister transfer lines will be provided in the Waste Handling Building. The lines will be operated concurrently to handle canistered waste transfer throughput. Each cask preparation area will include an airlock, a cask preparation station, and a cask decontamination station.

Assembly Transfer System. The assembly transfer system will be located in the Waste Handling Building. The system will receive and cool shipping casks in preparation for handling in the pool. In the pool, spent nuclear fuel assemblies will be removed from the casks and nondisposable canisters will be cut open. The spent nuclear fuel assemblies will be transferred into baskets and stored or will be sent to the dryers, after which the assemblies will be loaded into disposal containers.

Waste Package and Disposal Container Remediation System. The waste package remediation system will receive retrieved waste packages that were damaged from the disposal container handling system and perform operations required to repair the containers. The system will be located in the Waste Handling Building.

Disposal Container Handling System. The system will prepare empty disposal containers for loading, receive full disposal containers from the assembly transfer system and canister transfer system, weld, test, and inspect the containers, and transfer them to the waste emplacement system. The system will be located in the Waste Handling Building and will include the areas for empty container preparation, welding, waste package staging, tilting, decontamination, transporter loading, operating galleries, and equipment maintenance. The waste packages will be prepared and loaded onto the transporter within a shielded cell that will be supported by a remotely operated horizontal lifting system, decontamination equipment, manipulators, and a horizontal transfer cart.

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2. DESIGN PROCESS

2.1 GENERAL DESIGN PROCESS

This section describes the progression of the repository and waste package designs, as well as significant elements of the design process itself. These elements include the decision-making and decision-documenting process and the quality assurance requirements imposed on the design.

2.1.1 Evolution of the Repository Design

2.1.1.1 Site Characterization Plan Conceptual Design

The *Site Characterization Plan - Conceptual Design Report* (SNL 1984) describes the first complete conceptual design of the repository, now referred to as the Monitored Geologic Repository. The design is summarized also in the *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988b, Chapter 6, Volume III, Part A). This conceptual design served as a basis for guiding the site characterization plan before conducting extensive exploratory work. The objective was to collect site characterization data without compromising the waste isolating capabilities of the site.

The *Site Characterization Plan Overview* further described features of this first conceptual design, which included relatively small-capacity, single-material waste packages with thin walls. These waste packages were designed for disposal of intact fuel assemblies (Phase l), consolidated spent nuclear fuel (Phase 2), and high-level radioactive waste. The surface waste handling facilities were likewise divided into a Phase l building that produced the intact assembly packages and a Phase 2 facility in which spent nuclear fuel rods were consolidated and packaged for disposal (DOE 1988a, pp. 34–38).

As described in the *Site Characterization Plan Overview*, drilling and blasting were the primary methods proposed for excavating the repository. The main ramp accesses and main drifts were the only subsurface features planned to be mechanically excavated. All waste packages were to be hauled underground by diesel-powered equipment. The access ramp gradients were relatively steep; the waste transport ramp was just under 9 percent and the mined rock ramp was slightly under 18 percent. Gradients for the subsurface main drifts and emplacement panels ranged from 4 to 6 percent. The waste packages were to be placed in vertical boreholes drilled into the floors of the emplacement drifts (DOE 1988a, pp. 38–42).

2.1.1.2 Important Design and Construction Decisions Subsequent to the Site Characterization Plan

After the *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada* was published in 1988 (DOE 1988b), several significant design- and construction-related decisions led to the repository and waste package designs presented in this document as the VA reference design.

Use of Mechanical Excavation Methods. In the *First Report to the U.S. Congress and the U.S. Secretary of Energy*, the Nuclear Waste Technical Review Board recommended that mechanical excavation methods be used to construct the repository (NWTRB 1990). This recommendation reflected the Board's desire to minimize disturbance of the host rock and to produce a more cost-effective design (NWTRB 1990, p. 33). This recommendation led the design to emphasize the development of long, parallel emplacement drifts and main drifts with both straight and slightly winding alignments. Curves were widened and sharp corners were eliminated because the tunnel boring machines could not make sharp turns.

Inclusion of Multipurpose Canister Transfer Capability. The second significant development was the introduction of the multi-purpose canister concept in 1992 and its adoption by the Office of Civilian Radioactive Waste Management (OCRWM) in 1994 (CRWMS M&O 1996b, *Mined Geologic Disposal System Advanced Conceptual Design Report*, Volume II, pp. 8–50). Before the introduction of the multi-purpose canister, it was planned that commercial spent nuclear fuel would arrive at the repository in relatively small transport casks holding four to nine spent nuclear fuel assemblies. Individual assemblies would be transferred from the transport casks into waste packages one at a time. The multi-purpose canister, however, introduced the concept of transporting a larger number of fuel assemblies, either 21 pressurized-water reactor or 40 boiling-water reactor assemblies, in a single canister. The concept called for transferring the loaded canister directly into a waste package if the canister was approved for such use by NRC during the licensing process (CRWMS M&O 1996b, *Mined Geologic Disposal System Advanced Conceptual Design Report*, Volume II, p. 8-56). This concept eliminated the need to handle individual spent nuclear fuel assemblies received at the repository in such canisters. The multi-purpose canister package is also much heavier and has a much higher heat output than the previous small waste package design (CRWMS M&O 1996b, *Mined Geologic Disposal System Advanced Conceptual Design Report*, Volume II, pp. 8-56, 8-57). Two dry canister transfer cells are included in the VA reference design to handle disposable canisters.

Elimination of Rod Consolidation. A major change in the concept of waste handling and design requirements of the surface waste handling facilities was the decision not to pursue rod consolidation. Rod consolidation involves the disassembly of the individual spent nuclear fuel rods from the assemblies. The fuel rods are removed from the matrix hardware that holds them at the proper spacing within the assembly. Theoretically, this allows the rods to be bundled together in a smaller space. However, industry tests with rod consolidation at a few reactor sites, pursuant to 10 CFR 50.59, were viewed by the utilities as disappointing, particularly with regard to compaction ratios. Also, the irradiated hardware must be handled, packaged, and disposed of, and the process of taking the assemblies apart creates additional radiological concerns with resulting cleanup requirements. The decision not to pursue rod consolidation meant that the disposal container and waste handling facility designs would incorporate only the handling and disposal of intact spent nuclear fuel assemblies. This is seen as a simpler and cleaner operation. While the repository may receive a small amount of spent nuclear fuel consolidated at a few reactor sites, the rods will be received in sealed containers that can be accommodated by the reference design without the need for special handling equipment.

Development of Thick-Walled Multibarrier Waste Packages. The design philosophy for the waste package changed in 1993 with the start of the advanced conceptual design and development of a waste containment strategy. The design focused on the concept of maximizing the amount of fuel in each waste package and developing a robust waste package design that could be demonstrated through analysis to have a very long life. Maximizing the amount of fuel per package minimizes the number of containers that must be purchased, loaded, sealed, tested, and emplaced. Developing the thick-walled, multi-barrier container with different degradation characteristics for each layer of wall material provides a robust engineered barrier design that works in concert with the characteristics of the natural barrier in the unsaturated zone to provide long-term containment of the wastes within the waste package.

Surface Waste Handling Systems Design. The surface waste handling concepts have evolved significantly. The elimination of rod consolidation simplified the waste handling process. The surface waste handling design presented in the advanced conceptual design involved a "dry" process in which fuel assemblies were removed from the transportation casks in a fuel assembly transfer cell and placed directly into the disposal container. Since the advanced conceptual design, an evaluation of this handling process has resulted in the addition of pools for transport cask unloading. Fuel assemblies are removed from the transportation cask underwater in a pool similar to those licensed by NRC and used at nuclear power plants. This "wet" handling process uses water for radiological shielding, contamination control, and heat dissipation. Unloading fuel in pools also provides greater flexibility in handling casks of different design that might evolve in the future. The pools also serve to facilitate recovery operations that might be necessary with regard to the spent nuclear fuel and/or the transportation casks. After unloading in the pool, the fuel assemblies are transferred to a fuel assembly transfer cell for drying and subsequent insertion into disposal containers.

Waste Package Design. The waste package described in the site characterization plan was a relatively thin-walled (5/8 in. thick) single material container. The current container is much thicker (4.8 in.) and consists of two distinctly different metals. Two lids, an inner and an outer, must be welded separately onto each container. The thickness of the outer barrier requires a large amount of weld material to seal the container. The long welding time required for each container, coupled with the requirement that the welding be performed and inspected by remote means, makes this process a significant design feature.

Repository Subsurface Design. The repository subsurface layout evolved through several configurations in the early 1990s. The *Exploratory Studies Facility Alternatives Study; Final Report* (SNL 1991) produced 34 layout options for evaluation. The result of a parallel effort investigating the risks and benefits of exploring the Calico Hills units was also integrated into the *Exploratory Studies Facility Alternatives Study; Final Report.* Although this study focused on various concepts for the exploratory facility, it also addressed the requirements of 10 CFR 60.21(c)(1)(ii)(d). This regulation requires DOE to evaluate options for repository features that are important to waste isolation. Option 30 emerged as the preferred option. Option 30 consisted of two access ramps connected by a main drift at the repository level, a test area near the south end of the block, and additional exploratory drifting in the underlying Calico Hills geologic unit (SNL 1991, Volume I, Table 2).

Shortly after the *Exploratory Studies Facility Alternatives Study; Final Report* was released, the Option 30 concept was slightly modified to show the location of the testing area at the north end of the block. The modified concept, which is described in *Documentation of the Evaluation of Findings of the ESF Alternatives Study Used to Develop a Reference Design Concept* (YMP 1991), was used as guidance for initiating Title I (preliminary) design of the Exploratory Studies Facility.

The design of the Exploratory Studies Facility proceeded on the basis of the modified Option 30 into fiscal year 1993. The advanced conceptual design of the Monitored Geologic Repository began in fiscal year 1993. Development of the multi-purpose canister concept also was underway at this time. In 1993, the repository designers determined that it would be difficult to develop a subsurface waste transportation system for these large, heavy waste packages based on conventional rail and transporter design because of the steep ramps planned for the Exploratory Studies Facility. Because these ramps are intended to be the primary accesses to the repository, the Exploratory Studies Facility design was modified to eliminate steep ramps. The rationale for this design is documented in the *Description and Rationale for Enhancement to the Baseline ESF Configuration* (CRWMS M&O 1993).

Another reason for changing the design was to reduce the cost of the underground transport system by making the ramps and main drifts compatible with a standard rail haulage system. Avoiding Type I faults and achieving greater separation between the repository and the saturated zone also were important factors. The repository-level configuration described in *Description and Rationale for Enhancement to the Baseline ESF Configuration* (CRWMS M&O 1993, Figure 2, p. 9) formed the basis for the Exploratory Studies Facility tunnel.

The evolution of the subsurface repository design continued with the release of the *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996b). This layout was based on the revised Exploratory Studies Facility layout and incorporated low-gradient ramps and main drifts. The emplacement drifts were essentially flat, with only a slight gradient down to the main drifts to promote gravity drainage of any water seepage into the emplacement drifts (CRWMS M&O 1996b, Volume II, p. 8-123). The waste packages were to be placed in two emplacement-drift "blocks," a larger upper block, located at the level of the existing main drift, and a smaller lower block, located east of the upper block and about 65–70 m (213–229 ft) lower (CRWMS M&O 1996b, Volume II, p. 8-77).

The subsurface repository design described in this VA document represents a more optimal and cost effective version of the design presented in the *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996b). The most significant changes include the following:

- Eliminates the need for and cost of excavating the lower block by developing a more efficient layout in the upper block and incorporating a slightly different waste emplacement philosophy
- Uses an emplacement gantry to place waste packages on steel supports supported by concrete piers within the emplacement drifts instead of using more expensive rail cars for such support
- Eliminates a 9-m-diameter, 4,000-m-long (30-ft-diameter, 2.5-mile-long) drift and associated costs by changing the proposed method for launching the tunnel boring machines
- Relocates the exhaust main below the level of the emplacement drifts so that the exhaust main can not serve as a pathway for water seepage into the emplacement drifts

The focus supporting the VA reference design is the repository safety strategy, discussed in Volume 4, which provides the focus for future design work leading to preparation of a license application.

2.1.2 Decision-Making in the Design Process

The design process, particularly in the conceptual and preliminary design phases, involves the following:

- Establish required functions
- Establish acceptance criteria, requirements and constraints, and interfaces with other systems
- Identify options and concepts for a design solution
- Evaluate quantitatively the degree of feasibility, functionality, cost and efficiency for each option and design concept
- Select option and design concept and document decision

As part of the design control process, the rationale for selecting the preferred option is documented in technical reports or design analyses, and the selected option is depicted visually on engineering OCRWM NRC-approved quality assurance program, these documents are subject to project-specific checking and review procedures to ensure that the decision bases are examined and evaluated by personnel other than the originator. These decision documents are released only after the quality assurance requirements for review and approval are satisfied. During the development of the conceptual design for the subsurface repository, the decision-making process ranged from very detailed and formal decision-making methods, such as those described in the *Exploratory Studies Facility Alternatives Study; Final Report* (SNL 1991, Volume II, Section 2), to simpler methods that involved subjectively ranking options against various criteria and selecting the preferred option based on the tabulation of numerical rankings.

2.1.3 Quality Assurance in the Design Process

This section discusses aspects of the implementation of an effective design control process within the overall quality assurance program as part of the NRC Key Technical Issue on Repository Design and Thermal-Mechanical Effects (NRC 1997c). A discussion of the key technical issue is located in Section 4.3.3.6 of Volume 4. The design control process has been the subject of several DOE-NRC interactions, and DOE considers the design control process adequate to support the development of the design for the LA.

The VA reference designs for the repository and waste package have been developed in accordance with a nuclear quality assurance program that complies with Subpart G of 10 CFR 60. 10 CFR 60.152 requires that DOE implement a program that complies with the criteria of Appendix B of 10 CFR 50. The DOE has complied with these requirements by preparing the *Quality Assurance Requirements and Description for the Civilian Radioactive Waste Management Program* (DOE 1998c), which contains information on each criterion in Appendix B of 10 CFR 50. DOE and its contractors have also developed individual procedures that must be followed to implement the project quality assurance program.

Quality assurance requirements are imposed only on those Yucca Mountain Project (YMP) activities that are important to radiological safety or waste isolation (quality-affecting). When a work activity is planned, it is evaluated for quality assurance program applicability. If the activity is quality-affecting, it is carried out in accordance with the quality assurance program. If it is a non-quality-affecting activity, it is executed using a parallel set of non-quality-affecting procedures.

The activity of preparing the VA was evaluated, as required, and was found not to be a quality-affecting activity. The basis for this decision is that the VA represents a compilation and summary of other bodies of work. Individual pieces of work that are quality-affecting were performed in accordance with applicable quality assurance requirements and procedures. The VA, however, in its capacity as a summary level informational document, does not fulfill a quality-affecting function. Quality-affecting information cited in the VA is traceable back to the referenced originating document.

2.2 NUCLEAR SAFETY ANALYSES

Engineering designs are evaluated to determine their efficacy. The designs are subdivided into structures, systems, and components. Certain structures, systems, and components perform functions related to nuclear safety while others do not. Strict quality assurance standards are applied to structures, systems, and components that support functions important to radiological safety. Analyses are performed for structures, systems, and components whose failure, from a credible event, could compromise radiological safety or waste containment. The results of these analyses are evaluated. Nuclear safety requirements and analyses are discussed in Volume 4, Section 2.

Nuclear safety applies to the Monitored Geologic Repository during the preclosure phase, and waste isolation applies to the postclosure period. For the preclosure phase, nuclear safety is generally considered to address radiological protection of workers and the general public and is discussed in terms of safety analyses for design basis events. For the postclosure period, waste isolation is considered to address radioactive releases to the biosphere and the results of analyses are termed performance assessments. Postclosure performance assessments are discussed in Volume 3.

2.2.1 Nuclear Safety Analyses During the Preclosure Period

Design basis events are defined by 10 CFR 60 as those natural and human-induced events that may occur before permanent closure of the facility. Analyses of design basis events are used to quantitatively define and evaluate the ability of the repository system and its components to protect the offsite general public and worker health and safety from radioactive waste according to applicable regulatory criteria. These analyses are used to evaluate preclosure radiological safety by identifying structures, systems, and components that are important to radiological safety and the required mitigating functions.

Analyses are ongoing to define and evaluate repository preclosure design basis events to identify structures, systems, and components that are important to radiological safety, required mitigating functions, and administrative limits (e.g., technical specifications). Design basis events are identified and analyzed to support the repository design schedule (i.e., limit or eliminate redundancy with other work elements and support near- and long-term schedule commitments). Wherever possible, existing methods and regulatory precedent have been identified and incorporated into the design effort.

The safety analyses of identified design basis events require that mathematical and analytical models of processes and events be developed, whether the design basis events are induced by nature or by the construction and operation of engineered components. The complexity of the interactions, as well as the unavoidable incompleteness of data, requires that uncertainty and variability be incorporated into each of the models.

The following sections discuss the work accomplished to assess the behavior of the repository operating systems in the preclosure phase, as well as plans to determine the adequacy of those analyses prior to submittal of the LA.

2.2.1.1 Design Basis Events

Once a design basis event is identified, its credibility is determined on the basis of frequency. Table 2-1 categorizes design basis events by frequency in accordance with 10 CFR 60.

If a design basis event is determined to be not credible, then no further analysis is required. If the event is determined to be credible, it is categorized in accordance with the 10 CFR 60 definition, and an analysis is performed to determine whether the dose limits associated with the applicable category can be met. Category 1 events (those with a frequency greater than or equal to one event every 100 years) require that the sum of annual doses, exposures, and releases not exceed limits specified in 10 CFR 20 for the public and repository workers. Category 2 events (those with a frequency less than one event every 100 years to greater than or equal to once every 1 million years) require that the consequences of a specific Category 2 design basis event not exceed dose limits for the public beyond the preclosure controlled area, as specified in 10 CFR 60. Seismic event frequencies are categorized differently from other design basis events, as determined by the *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997). Frequencies of once in 1,000 years and once in 10,000 years, respectively, are used to determine the Frequency-Category-1 and -2 design basis vibratory ground motions. The frequency of once in 1,000 years for Frequency-Category-1 design basis ground motion is a conservative translation of "those natural and human-induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area." The frequency of once in 10,000 years for Frequency-Category-2 design basis ground motion is appropriate and conservative based on the observations that (1) it is comparable to the frequencies of the design basis ground motions for operating nuclear power plants in the United States, (2) these accepted reactor design bases and their associated design acceptance criteria have resulted in acceptable safe seismic designs, (3) design acceptance criteria that are the same or comparable to those in reactor designs will be used in repository design, and (4) an operating

Table 2-1. Design Basis Event Frequency Categories

Frequency Category	Frequency	Design Basis Event Category Definition	Applicable Dose Limit (Total Effective Dose Equivalent)
1	Greater than or equal to once in 100 years	Those natural and human-induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area.	Worker: 5 rem/year Offsite: 0.1 rem/year
2	Less than once every 100 years and greater than or equal to once every 1 million years	Other natural and human-induced events that are considered unlikely, but are sufficiently likely to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety.	Offsite: 5 rem/event
NC [not credible]	Less than once every 1 million years	These events are deemed to be not credible. As a result, no quantitative dose limits are applied, and the events are not considered design basis events.	Not Applicable

monitored geologic repository is inherently less hazardous and less vulnerable to earthquake-initiated events than is an operating nuclear power reactor. Seismic event design basis ground motion with a frequency less than once in 10,000 years is deemed to be not credible. As a result, no quantitative dose limits are applied, and the events are not considered design basis ground motion. Seismic event frequencies used in design basis event analyses are intended for seismic design only and do not apply to other types of design basis events.

2.2.1.2 Design Basis Event Identification and Selection

The *Preliminary MGDS Hazards Analysis* (CRWMS M&O 1996d) was used to identify hazards with potential radiological consequences at the Monitored Geologic Repository. A comprehensive list of both natural phenomena and human-induced events was developed in the hazard analysis by systematically analyzing operating functions and potential energy sources that could interact to cause potential hazards. Natural phenomena were identified based on known or predicted geologic, seismologic, hydrologic, volcanic, and meteorological characteristics. Human-induced events include surface, subsurface, and airborne events. The hazards identified represent potential initiating events and design basis event scenarios for surface and subsurface repository operations which have occurred in the past, are ongoing, or could occur in the future, without regard to a specific site.

Of the events identified in the hazard analysis, those determined to be not applicable to the preclosure phase of a repository located at Yucca Mountain (e.g., avalanche, coastal erosion, dam failure, high lake level, high tide, or tsunami) do not warrant further evaluation because no conditions exist that enable the event to occur. The remaining events have been sorted first by source (externally initiated or internally initiated), and further into logical analysis groups. External events are those that occur as a result of natural phenomena (e.g., earthquake, flooding, lightning, and tornado). Internal events are those that are tailored to internal situations (i.e., collision, crushing, and fire). The events in each group were determined using such discriminators as type of initiating event, potential waste form interaction, or location.

The internal events were identified by synthesizing functional areas of design for the subsurface and surface system and potential interactions that could impact a radiological waste form (e.g., waste package, fuel assembly, and radiological waste processing system components). This list of events was generated by determining the applicability of generic internal events that could interact with the waste form and result in a radiological release. Each of these events will require, either individually or in combination with other events, one or more of the following analyses:

- Frequency analysis that categorizes whether or not the event is Category 1, Category 2, or beyond design basis
- Nuclear safety analysis that assesses whether a radiological release occurs as a result of credible events (i.e., Category 1 or Category 2 events)
- Consequence analysis that evaluates whether the radiological doses of the event are within regulatory requirements, and if they are not, identifies preventative or mitigative structures, systems, and components that ensure the radiological consequences are within regulatory requirements

Many of the events have been grouped into smaller sets of analyses because the result of a given event analysis will bound other events (e.g., a given consequence analysis need evaluate only the worst-case drop event, which will bound other discrete drop events). The evaluation of each design basis event has been prioritized in accordance with the design prioritization process described in Section 2.3 and scheduled to support the LA.

The analyses of design basis events will form the radiological safety basis for the preclosure operations of the Monitored Geologic Repository. Design basis event analyses that identify repository structures, systems, and components that play a role in preventing or mitigating radiological releases in one or more design basis event will be used in quality assurance classification analyses to identify the structures, systems, and components that are important to radiological safety. In these classification analyses, graded quality assurance controls are established according to the importance to radiological safety of the structures, systems, and components, as identified in the design basis event analyses.

2.2.1.3 Results to Date

Table 2-2 presents the design basis events under analysis, the current status of those analyses' relevant conclusions, and the impact of the conclusions.

2.2.1.4 Conclusions from Design Basis Event Analyses

The potential bounding design basis event for the repository preclosure period is a drop of a spent nuclear fuel basket in the surface waste handling facility (Event 6 of Table 2-2). Preliminary results from ongoing design basis event analyses indicate that, with the use of the high-efficiency particulate air filtration system provided in the VA reference design (see Section 4.2.4.2 of this volume), doses predicted from the Category 1 spent nuclear fuel assembly drop meet offsite dose limits at and beyond a 5-km (3.1-mile) preclosure controlled area boundary.

Potential bounding Category 2 design basis events include rockfall (Event 3), seismic (Event 4), canister drop (Event 7), and transporter transportation event (Event 8), scenarios. These events have the highest potential consequences in the event of the postulated design basis event scenario. Using modeling assumptions, results show that Category 2 regulatory dose limits can be met at and beyond a 5-km (3.1-mile) preclosure controlled area boundary. The radiological consequences of the remaining design basis events applicable to the Monitored Geologic Repository are less than those expected from the potential bounding events. The validity and the applicability of these preliminary results will continue to be scrutinized in the development of the preclosure radiological safety basis.

The design basis event analyses in progress have used statistical data and radionuclide release models to estimate scenario frequencies and radiological doses. Further analyses that incorporate evolving design features, operational constraints, and radionuclide release mechanisms will refine and may reduce further the calculated radiological consequences. Uncertainty assessments will identify the margins of compliance with radiological release limits and enable elimination of unnecessary design margin and conservatism to reduce the cost of protection systems in the design of the repository. Analyses in progress and other design basis event analyses that are required before design completion (e.g., criticality analyses), will be performed before the LA/Site Recommendation design is completed.

The design basis event analyses presented in this section demonstrate that preclosure operations of the Monitored Geologic Repository will be performed within the radiological dose limits of 10 CFR 60 and 10 CFR 20. Design activities to be completed before the LA that may affect design basis event-related analyses are described in Section 3.2 of Volume 4.

2.2.2 Waste Isolation During the Postclosure Period

Waste isolation in the postclosure period is based on combined performance of the natural and engineered systems, as described by the *Repository Safety Strategy: U.S. Department of Energy's Strategy to Protect Public Health and Safety After Closure of a Yucca Mountain Repository* (DOE 1998b). Engineering measures may be selected to control water contacting the waste packages, limit the rate of waste package corrosion, and retard transport of radionuclides from the waste packages through the engineered barrier system. These same waste package and engineered barrier system design features also serve other functions during the preclosure phase, including radiation shielding to limit adverse effects of radiolytic induced reactions, maintaining dry conditions during the thermal period, (i.e., the time period of high rock temperature in the region surrounding the emplacement drifts), and limiting the maximum temperature of the spent nuclear fuel

Table 2-2. Preclosure Design Basis Event Analysis Summary

Event Analyzed		Status of Analysis	Conclusion
1	Aircraft Crash	An analysis of aircraft events is ongoing.	Analysis results indicate the scenario frequency (i.e., aircraft crash into operations area where vulnerable radiological material is located) is less than one event per 1 million years, classifying this event as a beyond design basis event in accordance with 10 CFR 60.
2	Rainstorm-Related Events	An analysis of rainstorms, debris avalanching, flooding, and landslides is ongoing.	Underground facility entrances are above the probable maximum flood level. Although some radiologically controlled areas of the surface facilities may be located in the probable maximum flood zone, the floor level of these areas is designed to be above the flood level. Debris avalanching and landslides will not impact underground or surface facilities.
3	Rockfall/Ground Support Fall	An analysis has been completed to determine releases and consequences from an assumed rockfall within an open emplacement drift.	Dose requirements are met at a 5-km preclosure controlled area boundary without particulate filtration. The margin for meeting dose requirements increases at greater distances from the location of the radionuclide release due to dispersion.
4	Seismic Events	Frequency-category-1 and -2 earthquakes are currently being analyzed to determine the structures, systems, and components that must be able to perform their important to radiological safety function during and after a seismic event.	Frequency-category-1 and -2 earthquakes are those that are expected to occur once in 1,000 years and once in 10,000 years, respectively. Calculated consequences are compared to design basis event Category 1 dose consequences for seismic frequency-category-1 events, and Category 2 dose consequences for seismic frequency-category-2 events. Preliminary analyses have determined the structures, systems, and components that must maintain their important to radiological safety function during and after a frequency-category-1 or -2 seismic event.
5	Ashfall (from volcanoes)	The worst-case event currently being analyzed is an ashfall of 3 cm at the repository site.	The postulated scenario frequency that could produce a worst-case ashfall is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. Results from the ongoing analysis indicate that repository structures will not be impacted by the 3-cm ashfall loading. Note: An onsite volcanic event has been determined to be a beyond-design-basis event for the preclosure period. Results of volcanic activity during the postclosure period are presented in Volume 3, Section 4.4.2.
6	Spent Nuclear Fuel Assembly Events (i.e., drop, slapdown, drop onto sharp object, and collision)	The worst-case event currently being analyzed is a drop of a basket of four spent nuclear fuel assemblies onto another basket of 4 spent nuclear fuel assemblies with 8 assemblies damaged.	The calculated scenario frequency is greater than one event per 100 years, classifying this event as a Category 1 design basis event in accordance with 10 CFR 60. High-efficiency particulate air filtration will mitigate the dose to and offsite dose limits at the preclosure controlled area boundary.
7	Canister Transfer System Events (i.e., drop, slapdown, drop onto sharp object, and collision)	The worst-case event currently being analyzed is a drop of one vertically oriented vitrified high-level radioactive waste canister onto another canister. Both canisters are damaged equally. As radiological source terms become available, a drop event for DOE-owned spent nuclear fuel may be determined to be the bounding event for this system.	The calculated scenario frequency for a vitrified high-level radioactive waste canister is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. The dose requirements are met at a 5-km preclosure controlled area boundary without particulate filtration. If it is determined in the future that a drop of DOE-owned spent nuclear fuel is the bounding event for this system, high-efficiency particulate air filtration is expected to maintain offsite and worker doses within regulatory limits.

Table 2-2. Preclosure Design Basis Event Analysis Summary (Continued)

Event Analyzed		Status of Analysis	Conclusion
8	Transporter Transportation Event	The worst-case event is assumed to begin while the transporter is traveling down the North Ramp into the repository, impact a repository wall, and release radioactive particulates.	The calculated event frequency is less than one event per 100 years but greater than once in 1 million years, and is classified as a Category 2 design basis event in accordance with 10 CFR 60. Dose requirements are met at a 5-km preclosure controlled area boundary without particulate filtration.
9	Waste Package Handling System	The worst-case event currently being analyzed is a drop of one waste package from beyond its designed drop height.	The calculated scenario frequency is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. Dose requirements are met at a 5-km preclosure controlled area boundary without particulate filtration.
10	Shipping Cask Accident Study	The worst-case event currently being analyzed is a drop of one shipping cask.	The calculated scenario frequency is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. Shipping casks are licensed according to 10 CFR 71. Postulated normal conditions of transport and hypothetical accident conditions of 10 CFR 71 are expected to bound any design basis events postulated for Monitored Geologic Repository operations. Therefore, there is no expected radiological exposures associated with the receipt of shipping casks.
11	Tornado/High Wind	The worst-case event currently being analyzed is a tornado wind-driven missile impact to the Waste Handling Building.	Preliminary results indicate the calculated scenario frequency is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. The Waste Handling Building will be designed to withstand the effects of tornadoes and high winds, including wind-driven missile impacts.
12	Fire Hazards Analysis	The worst-case event currently being analyzed is a fire that spreads from a surface nonradiological area to a surface radiological area.	Preliminary results indicate the calculated scenario frequency is less than one event per 100 years but greater than once in 1 million years, classifying this event as a Category 2 design basis event in accordance with 10 CFR 60. Fire suppression system is designed to suppress a fire while meeting dose limits at the preclosure controlled area boundary.

inside the waste packages. The contributions of engineered structures, systems, and components to postclosure waste isolation are discussed in the following sections.

2.2.2.1 Control of Water Contacting the Waste Packages

The water that enters the repository waste-emplacement drifts will vary with location and time. During the thermal period, the penetration of heated water into the drifts will be impeded because the rock will be dry and the temperature will be greater than the boiling point of water near the drifts. Repository design and thermal loading will determine the extent and duration for which such conditions prevail and, also, the potential for drainage through the repository horizon within pillars between the drifts. During cooldown and the post-thermal period, there will be increasing likelihood of water seepage into the emplacement drifts. Control of water in the emplacement drifts is important because certain types of waste package corrosion may result from humidity. Transport of radionuclides from the waste packages will be very slow unless the waste is contacted by flowing water.

Ground support in the drifts, including the drift lining, will be designed primarily to prevent rockfall during the preclosure phase. Precast concrete (i.e., inverts) will form the drift floor, support the equipment used for waste package conveyance and inspection and drift closure operations. Ground support and invert components are unlikely to remain structurally intact beyond a few hundred years after closure. The drift lining will serve to control seepage for as long as it remains in place. After collapse and decomposition, lining material will still be present in chemical constituent form and can affect water movement within the drifts.

Engineered barrier system features such as backfill, drip shields, or a combination of both, are considered in the VA reference design as options. Their purpose is to delay liquid water from coming into contact with the waste package surface. These options are discussed in Section 5.3.

If significant seepage into the drifts occurs, the water inflow will come through natural fractures in the host rock. For this water to move back into the host rock and not accumulate in the emplacement drifts, it must also flow through fractures. For control of water contacting the waste packages, engineered measures such as those listed above could divert water that may impinge on the package from above. Other measures such as sumps and drain holes can be used if necessary to promote drainage of water from the drifts. The materials used for ground support, invert components, backfill, and other engineered barrier system features will be selected to preserve the functions of engineered measures designed to promote free drainage.

2.2.2.2 Waste Package Lifetime

The waste will be contained for as long as each waste package remains intact. The waste packages will consist of inner and outer barriers, designed so that the inner barrier will be protected from oxygen and moisture by the slowly corroding outer barrier during the thermal period when penetration of water into the emplacement drifts will be impeded because the rock temperature will be greater than the boiling point of water near the drifts. In addition, the outer barrier will provide radiation shielding during the first few hundred years to limit adverse effects from radiolytic production of chemically aggressive compounds on the surfaces of the packages.

Once water returns to the emplacement drift in the form of humidity or seepage, there is the potential for chemical interaction with introduced materials, such as steel or concrete, to produce chemical conditions conducive to waste package corrosion. The possible reactions will depend on the reactivity of the engineered materials and the host rock, while the potential consequences will depend on the abundance of the reactants and the proximity of the reaction products to the waste packages. The same chemical conditions that promote corrosion of the outer barrier will not generally cause significant corrosion of the inner barrier. Pitting and crevice corrosion mechanisms that could affect the inner barrier may actually occur because of micro-environments within pits or cracks, in which chemical conditions deviate substantially from the bulk conditions of the emplacement drift. A complete description of components and the design of the waste package is provided in Section 5.1 of this volume.

The time to initial waste package penetration, the rate of subsequent corrosion, and the consequences to waste isolation performance will be influenced by major features of the repository design. Thermal loading and drift spacing control the temperature rise in the emplacement drifts and the extent of dryout in the surrounding rock. These conditions affect the elapsed time to return to pre-repository hydrologic conditions. In numerical simulations, drift spacing also controls the tendency for condensate to drain through pillars rather than flowing toward the emplacement drifts.

2.2.2.3 Limited Rate of Radionuclide Release

After a package inner barrier is penetrated by mechanisms such as pitting and crevice corrosion, it will continue to provide partial containment over an extended period of time. Dissolution of radionuclides is a direct function of the surface area of the waste form exposed and the amount of water that contacts the waste. The presence of intact Zircaloy cladding on spent nuclear fuel will reduce significantly the surface area available for radionuclide release. (No credit is taken for stainless steel cladding in conducting performance assessments.) Thermal limits are placed on the performance of the cladding to prevent oxidation that otherwise could compromise the cladding, expose additional surface area, and accelerate radionuclide dissolution rates (see Sections 3.2.1 and 5). The chemical composition of seepage that enters a partially failed waste package will be affected by the engineered materials present in the near-field (i.e., area that has been significantly impacted by the excavation of the repository and the emplacement of the waste). The effect on chemical conditions inside the waste package will be slight at first, but will increase gradually with time as the extent of waste penetration increases.

Mobilization of radionuclides will result from chemical transformation of spent nuclear fuel. Aqueous dissolution will be limited by the intrinsic solubility of the radionuclide species, and the rate of dissolution will depend on the ingress of water and oxygen. Spent nuclear fuel components may also be transported as colloidal particles that can form directly from dissolved components, or originate as natural minerals or engineered materials in the emplacement drift. Thus, the materials used for ground support and control of water contacting waste packages may eventually contribute to radionuclide transport from the waste packages.

Once radionuclides are released to the emplacement drift as dissolved or colloidal species, transport through engineered materials is required for release to the biosphere. Materials that filter colloidal particles may contribute to retardation, but conditions conducive to filtration may inhibit free drainage of the drifts. Materials with high affinity for holding radionuclides, such as iron oxides, may contribute colloidal particles. Materials that alter the near-field geochemical environment, such as concrete, may promote chemical precipitation of radionuclides, but also contribute to corrosion of waste package materials. The capacity of the emplacement drift for radionuclide retardation will be minor relative to the inventory of radionuclides present. Therefore, control of long-term performance by engineered measures is most sensitive to water contacting the waste and transport mechanisms, such as colloidal particles that may be increased by the presence of engineered materials.

2.2.2.4 Conclusions from Postclosure Waste Isolation Analyses

After the engineered barrier features perform their prescribed preclosure functions (e.g., controlling rockfall), they will remain in the emplacement drift. Measures to control water contacting the waste packages during postclosure will also involve introduced materials. Materials such as steel and concrete are chemically reactive and can affect corrosion of waste package materials and will affect the composition of water that eventually contacts the waste inside breached waste packages. Once radionuclides are released from the packages, engineered materials will make a minor contribution to retardation, but also can contribute to colloidal modes of transport. The retardation effect could be enhanced by the addition of sorbers to the drift inverts (see Section 5.2.2).

2.3 PRIORITIZATION OF DESIGN ACTIVITIES (BINNING)

To focus the design effort on those elements that are considered important to radiological safety or waste isolation, a system for prioritizing design work was developed so that the Monitored Geologic Repository structures, systems, and components would be ranked consistently. Items important to radiological safety comprise the structures, systems, and components that provide reasonable assurance that high-level radioactive waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the preclosure radiological safety dose (exposure) requirements of 10 CFR 60. Items important to waste isolation refer to the natural and engineered barriers that will be relied upon for achieving the postclosure performance objectives specified in 10 CFR 60, Subpart E. Prioritization also includes other considerations beyond radiological safety or waste isolation; namely whether design precedent has been established for particular structures, systems, and components in previous licensing actions.

Prioritization of structures, systems, and components design work is established through the assignment of a bin number. The three bin categories are defined as follows:

Bin 1: Indicates structures, systems, and components expected to have no significant radiological safety or waste isolation function or significance. These structures, systems, and components require the lowest level of design detail.

Bin 2: Indicates structures, systems, and components expected to have a radiological safety or waste isolation function, or other items of regulatory interest, with significant regulatory precedent indicated. These structures, systems, and components require more design detail than Bin 1 structures, systems and components.

Bin 3: Indicates structures, systems, and components expected to have a radiological safety or waste isolation function or impact, or other items of regulatory interest, with no applicable regulatory precedent. These structures, systems, and components require the highest amount of design detail.

Determining whether structures, systems, and components are important to radiological safety or waste isolation is based on the latest quality assurance classification of the Monitored Geologic Repository structures, systems, and components (CRWMS M&O 1997b). For preclosure analyses, this classification, which is based on preliminary conclusions made in advance of the completion of the evaluation of the appropriate Monitored Geologic Repository design basis events, is also the basis for the latest revision to the Monitored Geologic Repository *Q-List* (DOE 1998a). The *Q-List* consists of permanent items determined by analysis or direct inclusion (i.e., conservatively concluded by DOE to be important at the Monitored Geologic Repository, without conducting an analysis) to be subject to mandatory quality assurance requirements. For postclosure analyses, the basis for design prioritization is the total system performance analyses and their accompanying sensitivity studies, which facilitate evolution of the importance of the various functions to waste isolation. Relating functions to structures, systems, and components enables assessment for waste isolation importance as part of the design prioritization process.

The structures, systems, and components included in the prioritization process are drawn from the reference design for the Monitored Geologic Repository. The list of structures, systems, and components and their assigned priority is updated and modified, as necessary, as the Monitored Geologic Repository design progresses. Design prioritization is a planning tool that may require evaluating in-progress and conceptual-level design documents. These design documents may include system description documents, structures, systems and components architecture, design basis event analyses, and quality assurance analyses. Prioritizing the structures, systems, and components is based on engineering and management judgment. In some cases, management judgment may apply a more conservative priority on the basis of the lack of regulatory precedent rather than radiological safety or waste isolation.

Subsurface

There are 17 major systems that make up the subsurface repository architecture. These include:

Subsurface Systems	Bin
Subsurface Facility System	3
Engineered Barrier System	3
Ground Control System	3
Subsurface Ventilation System	3
Waste Emplacement System	3
Subsurface Closure and Sealing System	3
Waste Retrieval System	3
Subsurface Central Control System	3
Performance Confirmation System	3
Subsurface Electrical Distribution System	2
Subsurface Compressed Air System	2
Subsurface Water Distribution System	2
Subsurface Safety and Monitoring System	2

Subsurface Water Collection/Removal System	2
Subsurface Fire Suppression System	2

Subsurface Systems (Continued)	**Bin**
Subsurface Development Transportation System	1
Subsurface Emplacement Transportation System	1

Waste Package

A number of waste package designs have been developed to accommodate the variability in waste forms, all of which have been designated Bin 3.

Surface

There are 22 major systems that make up the surface repository architecture. These include:

Surface Facilities Systems	**Bin**
Canister Transfer System	3
Assembly Transfer System	3
Waste Package Remediation System	3
Disposal Container Handling System	3
Carrier Preparation Building Material Handling System	2
Site Generated Radiological Waste Handling System	2
Waste Handling Building Ventilation System	2

Surface Facilities Systems (Continued)	**Bin**
Carrier/Cask Handling System	2
Central Command and Control Operations System	2
Waste Transfer Building Ventilation System	2
Waste Handling Building Radiological Monitoring System	2
Performance Confirmation Data Acquisition/Monitoring System	2
Waste Handling Building Fire Protection System	2
Site Communications System	2
Site Water System	2
Emergency Response System	2
Site Electrical Power System	2
Security and Safeguards System	2
Site Compressed Air System	2
Pool System	2
Administrative System	1
Carrier/Cask Transport System	1

3. DESIGN BASES

This section describes the design bases for Monitored Geologic Repository design. These design bases provide the information that identifies the specific functions to be performed by the Monitored Geologic Repository, as well as the specific values or ranges of values chosen as controlling parameters to bound the design. This section also describes the design requirements for the proposed repository, the primary assumptions governing design, the allocated postclosure functions, and the preclosure radiological goals and objectives.

3.1 DESIGN REQUIREMENTS

The statutory and regulatory requirements that govern the proposed repository design and concept of operations, which are discussed in the VA, include the following:

- Nuclear Waste Policy Act, 42 U.S.C. §10101 et seq.
- 10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories
- 10 CFR 20, Standards for Protection Against Radiation
- 10 CFR 73, Physical Protection of Plant and Material

These statutory and regulatory requirements have been evaluated and baseline requirements have been developed for the overall Civilian Radioactive Waste Management System (CRWMS) program and for the Monitored Geologic Repository. The CRWMS requirements are documented in the *Civilian Radioactive Waste Management System Requirements Document* (DOE 1996a), and the Monitored Geologic Repository requirements are documented in the *Mined Geologic Disposal System Requirements Document* (CRWMS M&O 1996b). The regulatory requirements discussed in this section are addressed in greater detail in these two requirements documents. The requirements documents will be updated, as necessary, to incorporate modifications necessitated by changes in the statutory and regulatory requirements.

Using the requirements documents (DOE 1996a; DOE 1997c) as the basis, the following two project-level requirements documents were developed to identify the requirements for the design of the repository and the engineered barrier systems; the *Repository Design Requirements Document* (YMP 1994b) and the *Engineered Barrier Design Requirements Document* (YMP 1994a). These project-level requirements documents include design, operation, and decommissioning requirements that impact the physical development of the repository. The interface requirements between the Monitored Geologic Repository and other CRWMS projects are also documented in these baseline requirements documents.

The remainder of this section summarizes the design requirements for the repository and engineered barrier system. These requirements have been grouped into the following five general areas:

- Waste receipt
- Containment and isolation
- Criticality
- Retrieval
- Service life

The requirements presented are the minimum requirements for designing the Monitored Geologic Repository. The assumptions actually used to guide the Monitored Geologic Repository design generally are more conservative (see Section 3.2).

3.1.1 Waste Receipt

The waste forms received at the Monitored Geologic Repository will be in solid form, and as discussed in the *Engineered Barrier Design Requirements Document*, any particulate waste will be consolidated before being shipped to the repository (YMP 1994a, Section 3.7.1.1.A and 3.7.1.1.C). The waste will not contain ignitable or chemically reactive materials that could compromise waste containment or isolation (YMP 1994a, Section 3.7.1.C). Neither the waste forms nor the disposable canisters will contain free liquids that could compromise waste containment (YMP 1994b, Section 3.7.1.D). The acronyms MTU and MTHM are equivalent for the once-through nuclear fuel cycle (no reprocessed fuel) and can be used interchangeably. For commercial fuel, metric tons of heavy metal (MTHM) is the total amount of heavy metal initially contained in nuclear reactor fuel. For reprocessed nuclear fuels, the initial heavy metal loading contains both uranium and plutonium. For once-through nuclear fuels, the initial heavy metal loading consists of only uranium. Thus, for the United States nuclear industry, which uses a once-through fuel cycle, metric tons heavy metal equals metric tons uranium (MTHM = MTU). Table 3-1 shows the amounts of waste to be received annually at the repository.

Table 3-1. Annual Repository Receipt Rates

Year	Commercial Spent Nuclear Fuel (MTU)	Commercial High-Level Radioactive Waste, Defense High-Level Radioactive Waste, and DOE-Owned Spent Nuclear Fuel (MTU)
2010	400	TBD
2011	600	TBD
2012	1,200	TBD
2013	2,000	TBD
2014	3,000	TBD
2015	3,000	400 (TBD)
---	3,000	400 (TBD)
---	3,000	400 (TBD)
---	3,000	400 (TBD)
2031	3,000	400 (TBD)
2032	3,000	200 (TBD)
2033	1,800	TBD

Source: YMP 1994b, 3.2.1.2.B
TBD–To be determined, based on early receipt of naval spent nuclear fuel and immobilized plutonium starting in 2010.
The values for Columns 2 and 3 for years 2016 through 2030 are 3,000 and 400 respectively for each year.

As discussed in the *Civilian Radioactive Waste Management System Requirements document* and the *Mined Geologic Disposal System Requirements Document*, the proposed repository is statutorily limited, through the issuance of a license by NRC, to emplacing no more than 70,000 MTU of spent nuclear fuel and high-level radioactive waste until a second repository is in operation (DOE 1997c, Sections 3.2.1.E and 3.2.1.C). The waste allocation has been divided as follows:

- 63,000 MTU commercial spent nuclear fuel
- 640 MTU equivalent commercial high-level radioactive waste
- 4,027 MTU equivalent defense high-level radioactive waste
- 2,333 MTU DOE-owned spent nuclear fuel

This waste, which will be located at least 200 m (656 ft) below the surface (DOE 1996a, Section 3.7.2.2.F), is limited to material not regulated under the Resource Conservation and Recovery Act (DOE 1997c, Section 3.2.1.F). In addition, the design of the proposed repository is based on receiving the majority of DOE-owned spent nuclear fuel in disposable canisters suitable for direct insertion into waste packages (DOE 1997c, Section 3.1.5.R).

3.1.2 Containment and Isolation

The purpose of the following requirements is to limit the overall release of radionuclides to the biosphere. These requirements apply to the total system, including the engineered barrier system (which includes the waste packages and the underground facility) and the natural barrier.

- The waste packages must provide substantially complete containment for a period of at least 300 to 1,000 years (10 CFR 60.113 (a)(1)(ii)(A), NRC Staff Position 60-001, YMP 1994a, Section 3.7.D and YMP 1994b, Section 3.2.3.2.2.A.11).
- The engineered barrier system must limit any release from the waste packages so that the release to the geologic setting is gradual, and only a small fraction is released over a long period of time (10 CFR 60.113(a)(1)(i), YMP 1994a, Section 3.7.C).

- The engineered barrier system will limit the release rate of any radionuclide to less than 1 part in 100,000 per year for the amount of that radionuclide that is calculated to be present 1,000 years following permanent closure of the repository (10 CFR 60.113 (a)(1)(ii)(B), YMP 1994a, Section 3.7.E).

Repository design integrates the design requirements with the following attributes of the *Repository Safety Strategy: U.S. Department of Energy's Strategy to Protect Public Health and Safety After Closure of a Yucca Mountain Repository* (DOE 1998b):

- Limited water contacting the waste packages
- Long waste package lifetime
- Slow rate of release of radionuclides from the breached waste package
- Radionuclide concentration reduction during transport through the engineered and natural barriers

Past assessments have demonstrated that these four attributes are the most important in developing the safety case for a potential repository located in unsaturated tuff at Yucca Mountain. See Volume 4, Section 2 for more information.

3.1.3 Criticality

For purposes of this discussion, criticality is a condition that occurs when a sufficient mass of fissionable material, in a supporting geometry and in the presence of a neutron moderator, achieves a self-sustaining energy release. For preclosure safety, the repository systems are designed in such a manner that the waste will not become critical unless at least two independent and unlikely events occur, either in sequence or concurrently. For design basis events, the nuclear criticality calculations will result in an effective neutron multiplication factor of less than 0.95 including all biases and uncertainties, thus ensuring a 5-percent subcritical margin (YMP 1994b, Section 3.2.2.5). For the postclosure phase, the repository systems are designed so that the probability of any criticality will be small and that the consequences of any criticality will not compromise the repository's ability to meet its performance objectives (see Section 4.4.4 of Volume 3).

3.1.4 Retrieval

The initial requirement for the repository design ensured that any and all waste could be retrieved up to 50 years after waste emplacement operations were initiated (YMP 1994b, Section 3.2.1.4.B). However, during development of the VA reference design, this period was extended to 100 years, as discussed in the Section 3.2. Retrieval of the waste packages will be accomplished using the equipment and processes described in Section 4.2.7 (YMP 1994b, Section 3.7.4.1.A.4).

3.1.5 Service Life

A conservative approach has been taken also with regard to the service life of the facility. To accommodate the 50-year retrieval period discussed previously, a service life of at least 100 years would be required (YMP 1994b, Section 3.2.5.4). However, to support the more conservative 100-year retrieval period, in accordance with the assumption in Section 3.2.1, the Monitored Geologic Repository has been designed with a maintainable service life of at least 150 years. The 150-year service life is for design purposes only, however, and not for cost estimating purposes.

3.2 PRIMARY ASSUMPTIONS

During the course of the design development, the requirements discussed in Section 3.1 have been interpreted, updated, and supported with assumptions developed by the CRWMS Management and Operating Contractor (M&O). These assumptions are documented and controlled in the *Controlled Design Assumptions* document (CRWMS M&O 1998b). This document contains the key assumptions that affect many areas of the VA reference design. The assumptions supplement the technical baseline requirements and provide quantified values for technical data. In addition, the *Controlled Design Assumptions* document identifies design concepts for the surface, subsurface, and waste package to ensure a completely integrated system. All assumptions must be justified prior to development of any license application. Resolution strategies have been developed and are being implemented as a management tool to track each assumption to resolution.

This section discusses the key assumptions that affect the VA reference design in the following areas:

- System-level design assumptions
- Assumptions affecting the waste receipt rates
- Assumptions related to criticality calculations

As supporting analyses are completed and baselined, the assumptions will be removed from the *Controlled Design Assumptions* (CRWMS M&O 1998b) document and integrated into the LA design.

3.2.1 System-Level Assumptions

A primary design consideration for the repository is the effect of the repository on the waste and the natural environment. The following general assumptions have been made in the VA reference design to limit these impacts:

- The surface, subsurface, and waste package designs will be based on a reference areal mass loading range of 80 to 100 MTU/acre (CRWMS M&O 1998b, Key 019).
- The change to the ground temperature directly above the repository at a depth of 45 cm (17.7 in.) will be limited to no more than a 2C° (3.6F°) rise above the naturally occurring variability at that depth (CRWMS M&O 1998b, EBDRD 3.7.G.4).
- As a goal to limit the changes to the sorptive properties of the zeolite layer, the temperature at the top of the zeolite layer beneath the proposed emplacement area will not exceed 90°C (194°F) (CRWMS M&O 1998b, DCSS 025).
- The temperature of the emplacement drift wall will be limited to no more than 200°C (392°F) (CRWMS M&O 1998b, EBDRD 3.7.G.2).
- The temperature of the fuel cladding will be limited to 350°C (662°F) (CRWMS M&O 1998b, DCWP 001).
- For the VA design, the retrievability period identified in Section 3.1 has been increased to 100 years. Accordingly, the repository will be designed to allow for retrieval of any or all of the waste for as long as 100 years after initiation of waste emplacement operations (CRWMS M&O 1998b, Key 016).

If possible, the repository openings will be located to avoid Type I faults. For unavoidable Type I faults that intersect the emplacement drifts, a distance of 15 m (49 ft.) will be maintained from the edge of the fault to the nearest waste package (CRWMS M&O 1998b, Key 023). In addition, the design of the components important to radiological safety will be based on the method presented in the *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (CRWMS M&O 1998b, Key 064).

Finally, the repository design will not preclude the use of emplacement drift backfill at the end of the preclosure period (CRWMS M&O 1998b, Key 046), and the shafts, ramps, and exploratory boreholes will be permanently sealed upon closure of the repository (CRWMS M&O 1998b, Key 087).

3.2.2 Waste Receipt

Using the annual receipt rates presented in Section 3.1, it is assumed that the Monitored Geologic Repository will receive the following:

- 1,447 casks of spent nuclear fuel shipped by legal-weight truck

- 4,664 casks of uncanistered spent nuclear fuel shipped by rail
- 3,493 casks of canistered spent nuclear fuel shipped by rail
- 1,663 casks of commercial and defense high-level radioactive waste shipped by rail
- 1,203 casks of DOE-owned spent nuclear fuel shipped by rail

The annual cask receipt rates at the repository are provided in Key Assumption 001 of the *Controlled Design Assumptions* document (CRWMS M&O 1998b, Key 001). According to these cask arrival scenarios, Key Assumptions 002 and 003 provide the assumed waste arrival schedule and waste package emplacement schedule (CRWMS M&O 1998b, Key 002 and 003). In addition, it is assumed that only a small amount of DOE-owned spent nuclear fuel (approximately 50 MTU) will be received uncanistered (CRWMS M&O 1998b, Key 005).

In addition, the following assumptions have been made concerning commercial spent nuclear fuel received at the repository:

- Spent nuclear fuel assemblies from pressurized-water reactors will be, on average, 25.9 years out-of-reactor with a 3.69 weight percent initial enrichment and a burnup value of 39.56 gigawatt days/MTU.
- Spent nuclear fuel assemblies from boiling-water reactors will be, on average, 27.2 years out-of-reactor with 3.00 weight percent initial enrichment and a burnup value of 32.24 gigawatt days/MTU (CRWMS M&O 1998b, Key 004).

3.2.3 Criticality Calculations

The preclosure and postclosure criticality control period for the analysis method calculations, discussed in Section 3.1.3, is assumed to last for a period of 10,000 years or more (CRWMS M&O 1998b, Key 039). Calculations of the criticality of the repository waste during this period will take into consideration fission product credit, the principal isotope burnup in commercial light-water-reactor spent nuclear fuel (CRWMS M&O 1998b, Key 009) and the presence of neutron-absorber material in or from criticality control panels and rods made of long-term material (CRWMS M&O 1998b, Key 081).

3.3 ALLOCATED POSTCLOSURE FUNCTIONS

The primary postclosure function to contain the waste is described in the *Mined Geologic Disposal System Functional Analysis Document* (CRWMS M&O 1996c). This section identifies and describes the lower level functions that the Monitored Geologic Repository must perform to be successful in containing the waste from the accessible environment.

The function of containing the waste begins when the waste is sealed in the disposal container (waste package) (CRWMS M&O 1996c, p. 3-182, Function 1.4.5). The goal is to contain the waste in the waste package and inhibit the transport of radioactive material to the accessible environment. The natural barrier and the engineered barrier system (including the waste package) contribute toward meeting this overall function. This function is divided into the following subfunctions:

- Confine the waste
- Limit the radionuclide release to the natural barrier
- Limit the radionuclide release to the accessible environment
- Limit the natural and induced environmental effects

3.3.1 Confine the Waste

The function of confining the waste is performed by the waste package and includes limiting radionuclide release from the waste package by providing physical and chemical stability for the waste inside the waste package, and providing physical and chemical integrity for the waste package (CRWMS M&O 1996c, p. 3-183, Function 1.4.5.1). The function begins when the disposal container is sealed and continues until the waste package degrades to a state that provides no degree of confinement. The principal factors affecting this postclosure function are as follows:

- Chemistry of water on the waste package
- Integrity of the outer carbon steel waste package barrier
- Integrity of the inner corrosion-resistant waste package barrier

3.3.2 Limit the Radionuclide Release to the Natural Barrier

This function limits the movement of radionuclides from the engineered barrier system to the natural barrier (CRWMS M&O 1996c, p. 3-189, Function 1.4.5.2). The function is performed by the engineered barrier environment (thermal, hydrological, chemical, and mechanical) and the components of the engineered barrier system other than the waste package. Successful performance of this function slows the rate at which radionuclides are released into the natural barrier. The function begins once the first radionuclide is no longer contained by a waste package, either because it has left the waste package or because the waste package offers no confinement. The principal factors affecting this postclosure function are as follows:

- Seepage into the waste package
- Integrity of the spent nuclear fuel cladding
- Dissolution of uranium oxide and glass waste forms
- Solubility of Neptunium
- Formation of radionuclide-bearing colloids
- Transport through and out of the waste package

3.3.3 Limit the Radionuclide Release to the Accessible Environment

Radionuclide release to the accessible environment is limited by controlling the rate at which radionuclides are transported through the natural barrier (CRWMS M&O 1996c, p. 3-198, Function 1.4.5.5.3). This function affects the dose to which the public is exposed in the accessible environment. It begins when radionuclides enter the natural barrier. The principal factors affecting this postclosure function area as follows:

- Transport through the unsaturated zone
- Flow and transport in the saturated zone
- Dilution from pumping
- Biosphere dilution

3.3.4 Limit the Natural and Induced Environmental Effects

This function controls the impacts of the natural system on the engineered barrier system and the impacts of the engineered barrier system on the natural barrier (CRWMS M&O 1996c, p. 3-206, Function 1.4.5.5). The function begins when the waste is initially emplaced. The function is performed by the natural barrier, the engineered barrier, and the subsurface layout. The principal factors affecting this postclosure are as follows:

- Precipitation and infiltration into the mountain
- Precolation to depth
- Seepage into the drift
- Effects of heat and excavation on flow
- Dripping onto the waste package
- Humidity and temperature at the waste package

3.4 PRECLOSURE RADIOLOGICAL GOALS AND OBJECTIVES

The regulatory and repository system-level requirements for radiological safety during preclosure apply to all of the areas within the radiologically controlled area. In addition, a preclosure controlled area boundary, which encompasses the radiologically controlled area, will be established around the waste handling building with a radius of 5 km (3.1 miles) (CRWMS M&O 1998b, Key 071). This section summarizes the key preclosure radiological requirements from the *Repository Design Requirements Document* (YMP 1994b).

The overall design of the repository will ensure the protection of both the workers and the general public. This will be accomplished by complying with the following:

- The design will ensure that the radiation exposures, radiation levels, and release of radioactive materials to unrestricted areas will be maintained within the limits specified by 10 CFR 20 and applicable environmental standards for radioactivity as established by the U.S. Environmental Protection Agency (EPA) (YMP 1994b, Section 3.2.1.2.C).

- The repository facilities will be designed to operate so that the total effective dose equivalent from the licensed operation to individual members of the public does not exceed 0.1 rem in a year (YMP 1994b, Section 3.2.2.2.A) and the dose in any unrestricted area does not exceed 0.002 rem in any 1 hour from external sources from Category 1 design basis events (YMP 1994b, Section 3.2.2.2.C).

- The design and operations will include provisions for controlling doses so that the exposure dose limits specified in 10 CFR 20.1201 for occupational doses, and 10 CFR 20.1301 for individual members of the public, are not exceeded.

- The total effective dose equivalent for an individual located at or beyond the preclosure controlled area boundary will not exceed 5 rem from Category 2 design basis events (10 CFR 60.136).

- The repository will be designed and constructed to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable. As low as is reasonably achievable principles will be based on the applicable portions of NRC Regulatory Guides 8.8 and 8.10 (YMP 1994b, Section 3.2.2.1.A).

- Access to radiation areas will be controlled in accordance with the requirements specified in 10 CFR 20.1601 and 20.1602 (YMP 1994b, Section 3.2.4.3.2).

- The design will include equipment to monitor the external surfaces of waste packages and casks for radioactive contamination and radiation levels in compliance with 10 CFR 20.1906 (YMP 1994b, Section 3.2.4.4).

- The shielding design will limit the maximum exposure to an individual worker to one-fifth of the annual total effective dose equivalent limits for normally occupied areas as specified in 10 CFR 20.1201 (YMP 1994a, Section 3.2.4.5.1.A). The shielding will be designed with the goal of limiting the dose rate in intermittently occupied areas to less than 1 rem/year based on occupancy, time, and frequency of exposure (YMP 1994b, Section 3.2.4.5.1.B).

- Radiation monitoring devices will be designed to monitor individuals, equipment, and key areas. These monitors will include visual and audible alarm systems (YMP 1994b, Section 3.2.2.4).

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4. REPOSITORY DESIGN

Equipment and procedures for waste handling already approved by NRC and in use at other licensed nuclear facilities have been incorporated into the design and operating concepts to the maximum extent practicable.

Suggested guidance received from industry experts on the Repository Design Consulting Board has been incorporated into the design and concept of operations. Such guidance was provided in the areas of waste receipt, transfer, handling, and packaging; underground excavation, ventilation, and ground support; waste package design; and waste package materials testing. The waste package design and materials testing program are discussed in Section 5.

4.1 REPOSITORY SURFACE FACILITIES

This section provides detailed descriptions of the radiologically significant waste handling facilities, systems, and equipment located in the Radiologically Controlled Area. The controlled area refers to that surface area protected by fences and security systems where the radiological waste shipments will be handled and the site-generated low-level radioactive waste will be processed. During development of the VA, emphasis was placed on the design of the facilities, systems, and equipment that must handle the waste carriers, prepare the shipping casks for unloading the waste, and load the waste into disposal containers. This section also briefly covers some balance of plant support facilities and systems.

4.1.1 Surface Operations Overview

The primary mission of the surface facilities will be to receive spent nuclear fuel and high-level radioactive waste shipments and prepare and package the wastes for underground emplacement.

This section defines the functions that must be performed to package the waste and provides a summary description of surface waste handling activities during the emplacement phase. It focuses on the areas of the North Portal of the repository, as seen on Figure 4-1, North Portal Surface Facilities.

Figure 4-2, Waste Receiving and Shipping Operations, and Figure 4-3, Waste Handling Operations, depict a general overview of the path that waste will take through the surface facilities as it comes from a waste generator and is subsequently emplaced in the repository.

Waste shipments of commercial spent nuclear fuel constitute the majority of wastes that will be received at the repository. These shipments will be delivered to the repository North Portal area by the Regional Service Contractor using offsite prime movers. These prime movers are the locomotives and trucks that will be employed by the Regional Service Contractor to transport the waste by casks mounted on carrier rail cars and truck trailers to the repository from reactor sites. Defense wastes from government facilities may be delivered to the repository by other DOE transportation contractors. The offsite prime movers will wait outside the controlled area for dispatcher instructions. All waste shipments will be received at a security station, where they will be inspected. Casks and furnishings mounted on a carrier will be transported within the controlled area by a site prime mover. Waste shipments will be transported to the Carrier Preparation Building or to a parking area to wait for a bay in the Carrier Preparation Building.

At the Carrier Preparation Building, the personnel barriers and impact limiters will be removed and the cask, rail car, and/or trailer will be inspected. A contaminated cask can be cleaned by wiping at the Carrier Preparation Building or it can be moved to the Waste Handling Building for decontamination. The carriers can also be directed to a carrier washdown station to remove road grime.

The prepared carrier will be transported to the Waste Handling Building carrier bay, where the cask will be removed and the carrier will be loaded with an empty cask for shipment back to a waste generator. Systems in the Waste Handling Building will unload truck and rail shipping casks and send them to one of two waste handling systems: a wet assembly transfer system that includes a pool,

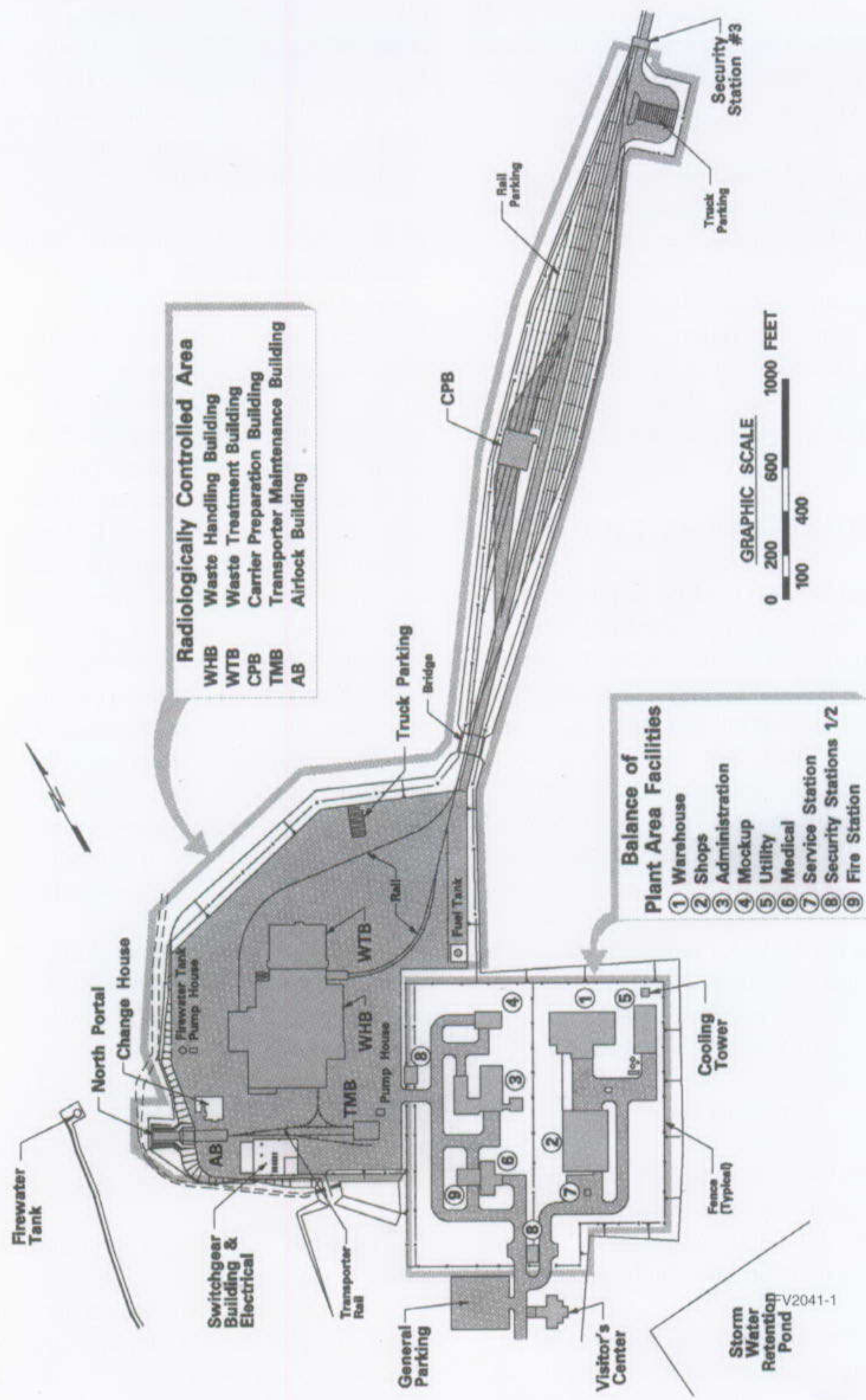

Figure 4-1. North Portal Surface Facilities

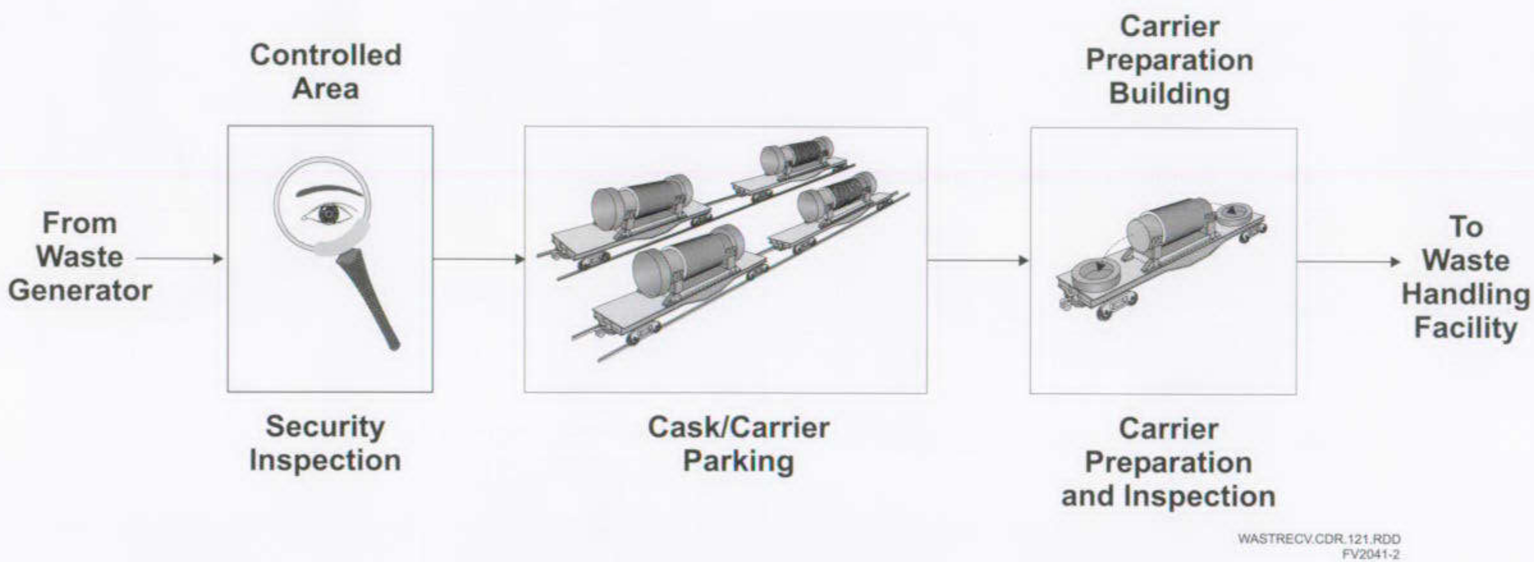

Figure 4-2. Waste Receiving and Shipping Operations

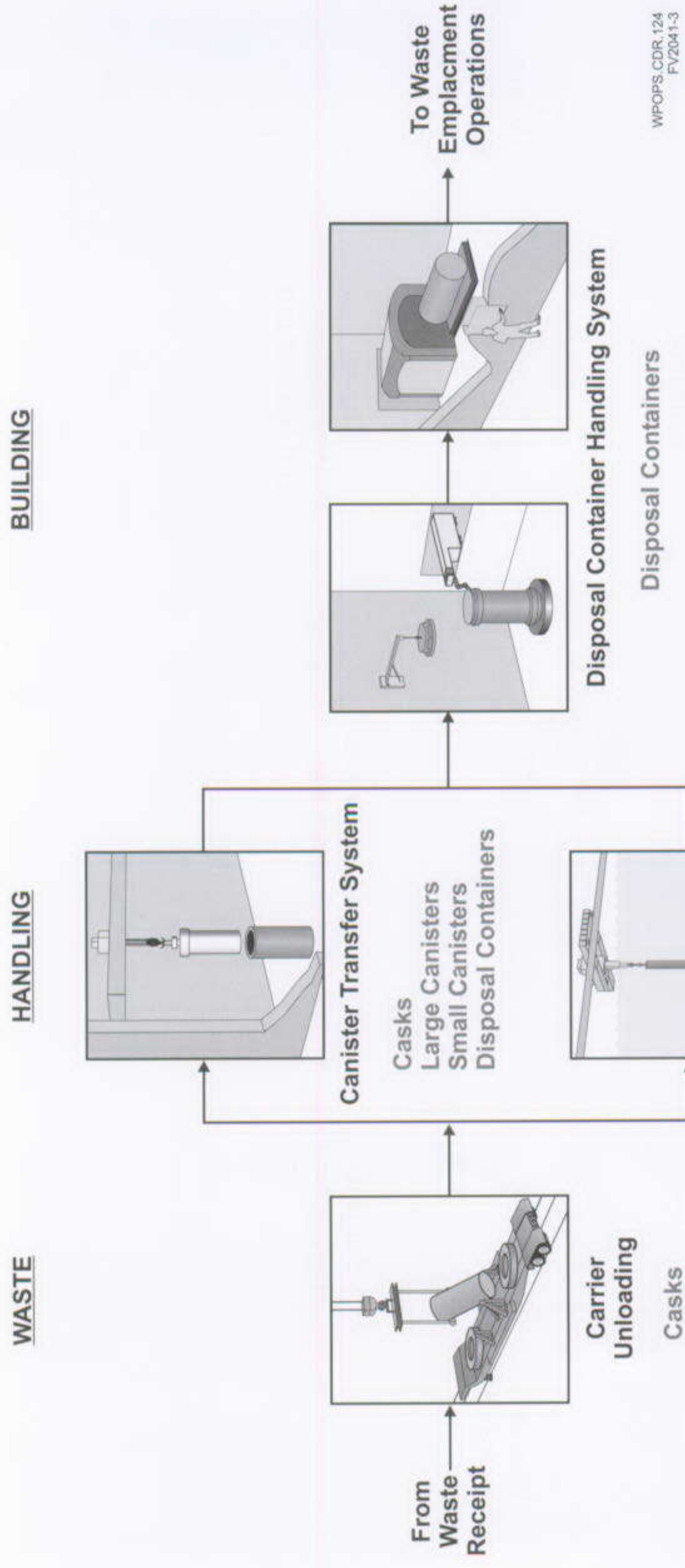

Figure 4-3. Waste Handling Operations

or a dry canister transfer system (for disposable canisters only).

The assembly transfer system will receive casks containing individual fuel assemblies that have either been loaded into the cask directly or are contained in a nondisposable canister that must be removed from the cask and opened before the assemblies can be removed. Some non-disposable canisters are welded closed and will be cut open. The assemblies will be removed from the casks or canisters in a pool environment, after which they will be transferred to, and dried in, a fuel assembly transfer cell prior to being loaded into a disposal container.

The dry canister transfer system will receive spent nuclear fuel, vitrified defense high-level radioactive waste, and special defense waste forms, including immobilized plutonium, in canisters designed for direct insertion into disposal containers.

The disposal container handling system will receive loaded containers from both the assembly transfer system and the canister transfer system. The inner and outer lids will be welded and inspected to ensure that they meet specifications for disposal. After the disposal container has been loaded, sealed, and tested, it is thereafter referred to as a waste package. The waste package will be turned to the horizontal position and loaded into a subsurface transporter, which takes it to an emplacement drift.

Empty casks will be loaded onto carriers in the carrier bay and returned to the Carrier Preparation Building, where impact limiters and personnel barriers will be re-installed. The cask/carrier will then be sent to a parking area or to the security station for offsite shipment. Empty nondisposable canisters will be placed into a protective overpack and prepared for shipment to an offsite facility for disposal or recycling. Arriving empty disposal containers will be delivered to the Security Station and from there to a parking area or empty disposal container preparation area in the Waste Handling Building.

Low-level radioactive waste and hazardous waste will be generated in the surface waste handling facilities and operating areas. Solid and liquid low-level radioactive waste will be safely accumulated at the point of origin, then sent to the waste treatment facility, where they will be treated as appropriate and packaged in drums. Hazardous waste will be collected and packaged in drums and sanitary waste will be collected for proper disposition. Low-level radioactive waste and hazardous waste will be shipped offsite to a licensed disposal facility. Mixed waste exhibits the characteristics of low-level radioactive and hazardous waste. It is not anticipated that mixed waste will be produced during waste handling operations, but there will be an allowance made for temporarily staging a small quantity of this waste prior to shipping it offsite.

4.1.2 Major Factors that Influenced the Surface Design

Two major factors influenced the design approach and operational concepts of the surface facilities: waste package size and weight, and waste receipt.

4.1.2.1 Waste Package Size and Weight

In the VA reference design, a large waste package will be emplaced horizontally in the subsurface emplacement drifts. As indicated in the *Preliminary List of Waste Package Designs for VA* (CRWMS M&O 1998o), spent nuclear fuel waste packages will generally hold 21 pressurized-water reactors or 44 boiling-water reactor spent fuel assemblies. The weight of the package, when loaded with waste and sealed, will be about 76 tons. The largest canistered waste form to be handled will be naval spent nuclear fuel in sealed, disposable canisters that weigh about 49 tons and have a total weight in the sealed waste package of 91 tons (CRWMS M&O 1998o). A discussion of disposable canisters later in this section provides more detail. Surface facility areas and support systems must be designed to minimize the adverse effects from either a cask or a waste package drop, including damage to facility safety systems. The design and specification of this equipment will require development and demonstration tests to confirm its operability, reliability, maintainability, and safety. The advantage of the larger waste packages will be the reduction of the number of disposal container units to be manufactured and the number of waste packages to be transported to the underground facility and emplaced. See Section 4.2.3.1, discussion on the Waste Package Transporter.

4.1.2.2 Waste Receipt

Spent nuclear fuel and high-level radioactive waste will arrive at the surface facilities in several modes. Transportation casks for commercial spent nuclear fuel may contain (1) uncanistered fuel assemblies or (2) a canister loaded with assemblies. Previously, a multi-purpose canister option was proposed that would have been licensed for dry storage, transportation, and subsequent disposal in the repository. This type of canister will now be referred to as a disposable canister since it can be placed directly into a disposal container for subsequent emplacement in the repository. Defense high-level radioactive waste forms and most DOE-owned spent nuclear fuel are expected to be received in disposable canisters. A small amount of DOE fuel, of commercial origin, will be uncanistered.

As noted in the *Controlled Design Assumptions Document* (CRWMS M&O 1998b), the current commercial spent nuclear fuel receipt configuration options are described in the following paragraphs.

Uncanistered Fuel Assemblies. Commercial spent nuclear fuel will be received in casks containing uncanistered fuel assemblies. The surface waste handling system design will be based on receiving approximately one-third of these assemblies in non-disposable (dual-purpose) canisters, and about two-thirds as uncanistered assemblies. Uncanistered fuel assemblies have the greatest impact on the complexity of the Waste Handling Building and associated waste handling and support systems. More handling stations are required to prepare and open the casks, and to transfer multiple fuel assemblies at staging, drying, and loading stations. In addition, more low-level radioactive waste is generated for handling at the waste handling and waste treatment facilities.

Disposable Canisters. Commercial spent nuclear fuel will be received in large casks containing one large disposable canister (a canister licensed for storage, transportation, and disposal). This form of receipt will have the least impact on the complexity of the surface waste handling facilities because transferring a sealed canister from a cask to a disposal container requires fewer intermediate steps and results in less waste and contamination. As stated before, naval spent nuclear fuel will also be received in disposable canisters, as will be some DOE-owned spent nuclear fuel.

Nondisposable Canisters. Commercial spent nuclear fuel will be received in large casks. Each cask contains one dual-purpose (non-disposable) canister, which will be licensed for storage and transportation only; that is, the canister is not licensed to be put in a disposal container and emplaced in the repository. The canister will be welded closed at the utility and, therefore, must be cut open at the repository so the spent fuel assemblies can be transferred into a disposal container. This option will have the greatest impact on the complexity of the design of the Waste Handling Facility because of the need for special equipment to cut open the canisters and subsequently transport the empty canisters offsite for disposal or recycling. The additional waste handling operations that will be required during canister preparation and decontamination create a considerable amount of low-level radioactive waste, especially during underwater operations, which include cutting the canisters open.

4.1.3 Carrier/Cask Transport System

The carrier/cask transport system will include the onsite roads and rail systems, parking areas, site prime movers, and equipment required to engage and disengage carriers from prime movers, park the carriers and prime movers, and transport the carriers inside the controlled area.

The system will receive casks at a peak annual rate of about 1,500 rail and 240 truck shipments, which will include waste forms, empty disposal containers, and nondisposable canister overpacks. Shipments will arrive at the security station and be inspected, and the offsite prime movers will be parked outside the gate. The carrier will be transported by a site prime mover to a parking area, the Carrier Preparation Building (described in the next section) or to the Waste Handling Building in the case of empty disposal container shipments.

The carrier washdown system will provide the equipment required to remove road grime from carriers. The carrier/cask transport system will provide the rail switching, safety systems, and truck turn-around required for carriers to be managed in and out of parking, the security gate, and the facility preparation and handling bays. A transporter maintenance facility will be located within the controlled area for maintaining the site prime movers for surface and subsurface operations. The system design, traffic control, and safety systems will be designed to U.S. Department of Transportation standards.

4.1.4 Waste Handling Facilities

The North Portal waste handling facilities will comprise a radiologically controlled area, which will include facilities associated with waste shipping, receiving, and preparation, and loading into disposal containers. Included in this section are descriptions of the primary facilities and the systems that will be associated with waste handling facilities. The facilities located in the controlled area will be dedicated to nuclear waste handling, shipping, and receiving operations, as described in the *Surface Nuclear Facilities Space Program Analysis* (CRWMS M&O 1997ak).

4.1.4.1 Carrier Preparation Building

The Carrier Preparation Building will house the material handling system required to prepare incoming carrier/cask configurations for unloading at the Waste Handling Building and for return shipment to waste generators (refer to Figure 4-4, Carrier Preparation Building). The structure will consist of steel framing with insulated metal siding and roofing. The building columns on spread footings will be designed to support the loads associated with two 10-ton bridge cranes.

The cranes will be used to remove and re-install the shipping cask impact limiters and personnel barriers. Continuous concrete mat foundations will be employed beneath the railroad tracks to support the heavy wheel loads. The facility will have a net floor area of approximately 20,000 ft^2, the majority of which will be allocated to four parallel roadways/tracks, each using recessed tracks. The facility will require no special features for radiation shielding since the radioactive waste will remain in the sealed transportation casks, which provide the necessary personnel shielding. Included will be the utilities, support, and safety systems required to support carrier/cask operations and protect personnel. The ventilation system will use overhead and/or wall-mounted radiant heaters, with cooling provided from roof-mounted fans.

Carrier/Cask Preparation System. The carrier/cask transport system will move rail and truck cask configurations to and from the Carrier Preparation Building, the security gate, and the Waste Handling Building. Carrier preparation operations will include moving a loaded carrier/cask ensemble into an available preparation bay with a site prime mover, performing radiation surveys, removing the personnel barrier, inspecting for contamination, measuring the cask temperature, and removing the impact limiters. The prepared carrier/cask can be transported to the Waste Handling Building when a position is available, or sent to a parking area.

Truck or rail carriers with empty casks will be prepared for offsite shipment by installing impact limiters, performing cask surveys, and installing personnel barriers. Four parallel preparation lines will operate concurrently to handle the waste within established shipment schedules. Each line has two preparation bays, each of which can accommodate truck and rail shipping and receiving. One remotely operated overhead bridge crane and gantry crane will be provided for each pair of lines, servicing four preparation bays. Remotely operated support equipment will include a gantry-mounted manipulator and the tooling and fixtures required for removing or installing personnel barriers and impact limiters. Remote handling equipment will be designed for operator safety, to minimize radiation exposure and facilitate maintenance.

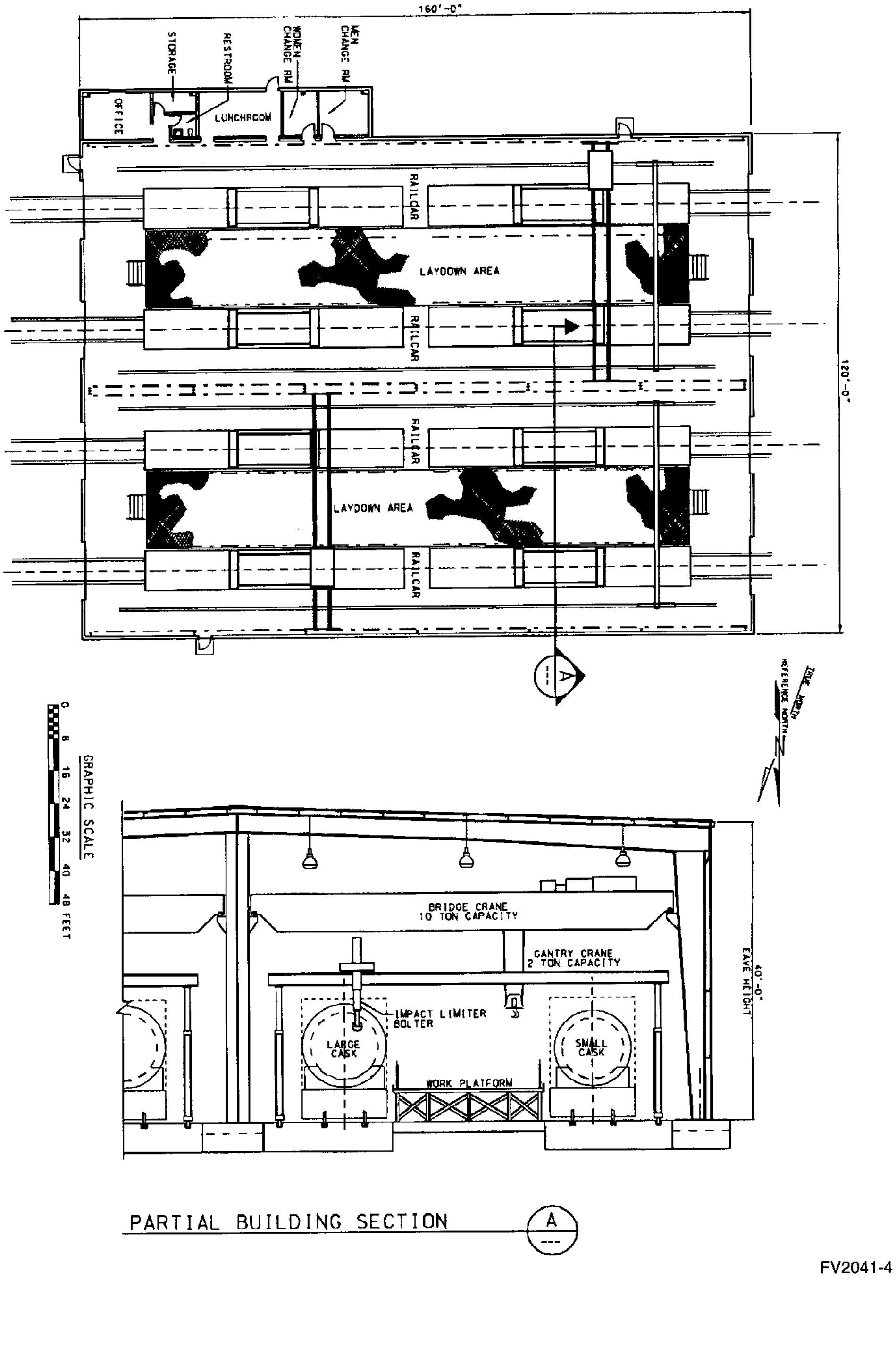

Figure 4-4. Carrier Preparation Building

4.1.4.2 Waste Handling Building

The Waste Handling Building will provide the structures, controlled areas, and accesses required to house and operate the waste handling systems, protect operating personnel, and maintain radiological confinement. Integral to the facility structure will be the essential waste handling systems, including the carrier/cask handling system, assembly transfer system, canister transfer system, disposal container handling system, and waste package remediation system. Essential facility support systems will include the facility electrical, fire protection, radiation monitoring, and ventilation systems. The *Surface Nuclear Facilities Space Program Analysis* describes the ancillary support systems, including those required for facility security, communications, alarm and public address, potable and chilled water, sanitary waste, low-level radioactive waste, effluents, air and vacuum, decontamination, waste handling pools, repair and calibration shop, offices, and facility monitoring and control systems (CRWMS M&O 1997ak).

Waste Handling Building Layout. The Waste Handling Building will be located close to the North Portal, within the controlled area. The structure will establish the operating and equipment areas; the boundaries required for safe handling of shipping casks, waste forms, facility waste, and disposal containers; and facility office and support operations (refer to Figure 4-5, Waste Handling/Waste Treatment Building).

The structure will have 440,000 net ft^2 of (useable) operating area and 462,000 gross ft^2 total area (including walls). There will be two floors located below grade and four above grade. The facility will have a variety of floor areas and area heights that range from 9 ft to over 120 ft above grade, with the taller heights required for the waste handling pools, high vertical lift areas, and equipment transfer corridors.

Building circulation pathways will accommodate the maximum number of facility personnel, including increased personnel present during shift changes. Fire egress pathways will be provided to meet the requirements of the *National Fire Protection Association's Code for Safety to Life from Fire in Buildings and Structures* (NFPA 1997). Hatches, structural and shielding walls, and a ventilation system will be provided to meet NRC requirements for radiological safety.

Waste Handling Building Operating and Equipment Areas. The Waste Handling Building operating and equipment areas and rooms are identified in Figures 4-6, Waste Handling Building Sections, and 4-7, Waste Handling Building Floor Plan. The carrier bay area will include two rail/truck lines into the bay and a carrier/cask handling system required for cask receipt and shipping. The assembly transfer area will include three waste handling lines with an assembly transfer system in each. Each line will include a dry transfer cell for cask preparation; a pool for cask and canister opening, fuel assembly unloading and staging; and a fuel assembly transfer cell for fuel assembly drying and disposal container loading. The canister transfer area will include two waste handling lines with a canister transfer system in each.

The disposal container handling system will contain a material handling system, including empty container preparation and disposal container handling cells for container welding, inspection and staging, decontamination, and transfer. The waste handling systems for these areas are described in detail in subsequent sections.

The primary support areas of the facility will include the operating galleries, equipment transfer corridors, and contaminated equipment rooms. The locations of operating galleries and control rooms will be designed to optimize waste handling system control and viewing and personnel movement. Lay down areas will be provided for cask components, handling fixtures, and tooling as required during normal and off-normal operations. To support system maintenance, hot and cold equipment servicing rooms and transfer corridors will be located near waste handling areas.

Pool support areas will contain the support equipment required to maintain water quality, including temperature and filtration, in three fuel assembly transfer pools associated with the assembly transfer

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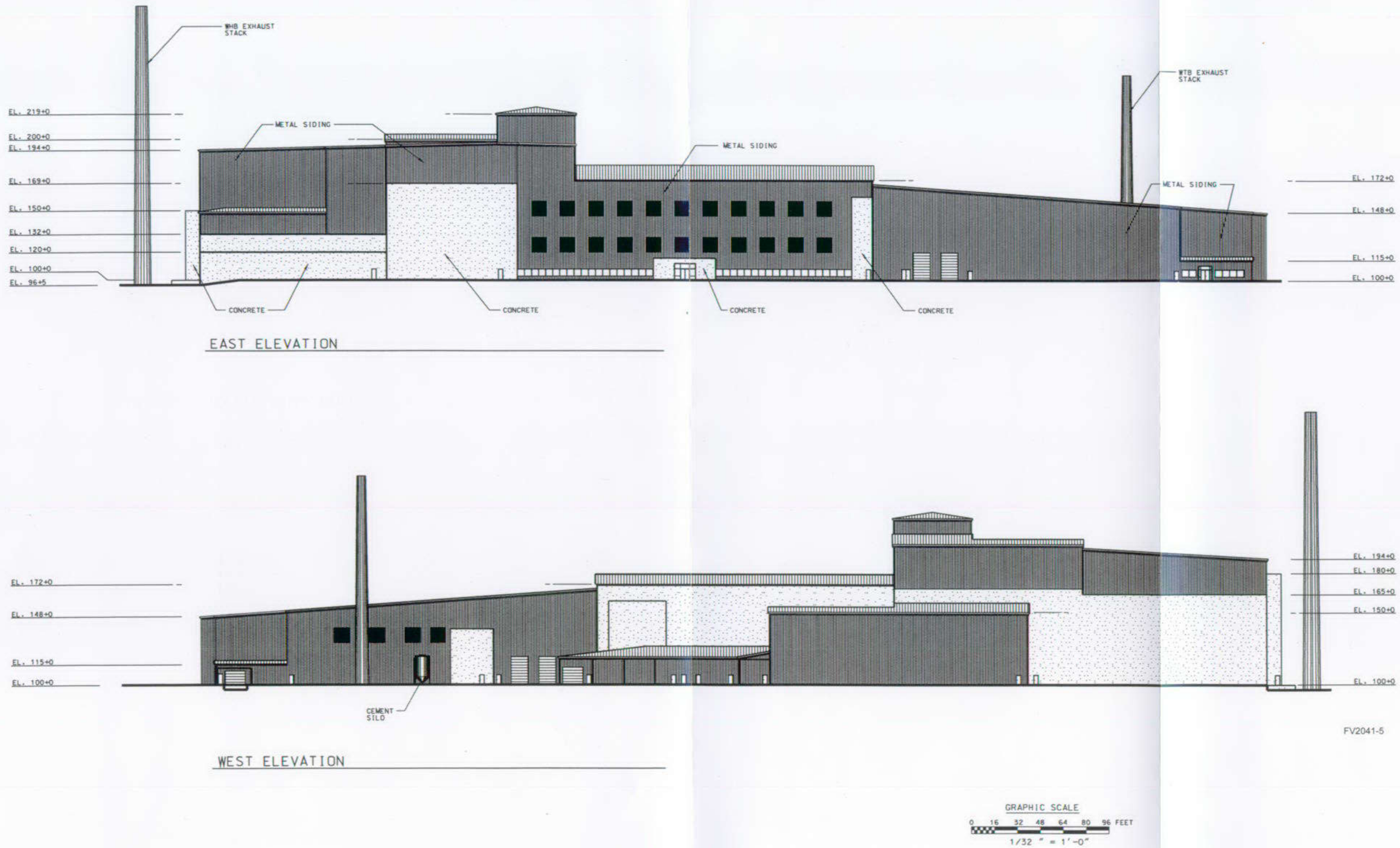

Figure 4-5. Waste Handling/Waste Treatment Building

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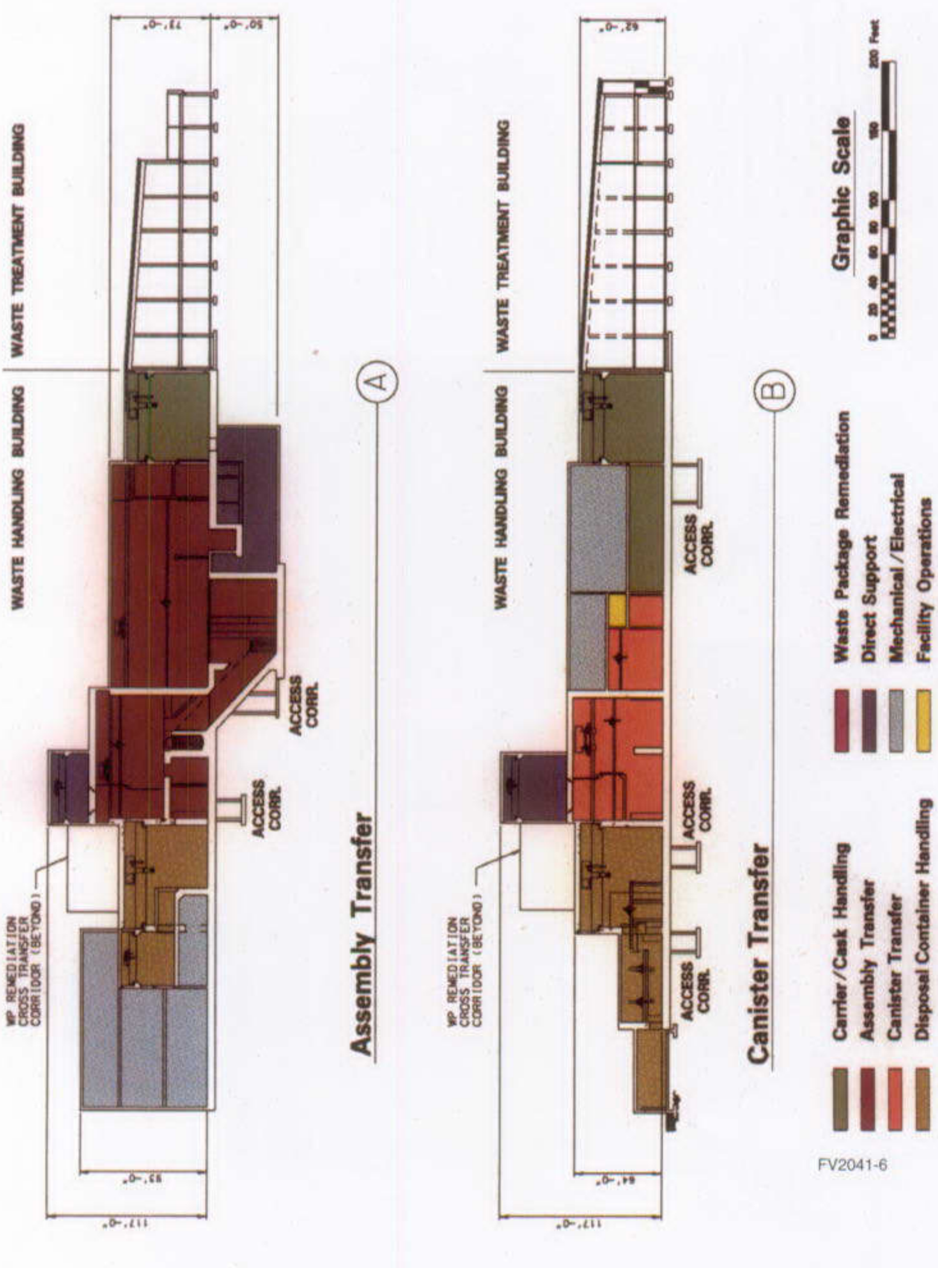


Figure 4-6. Waste Handling Building Sections

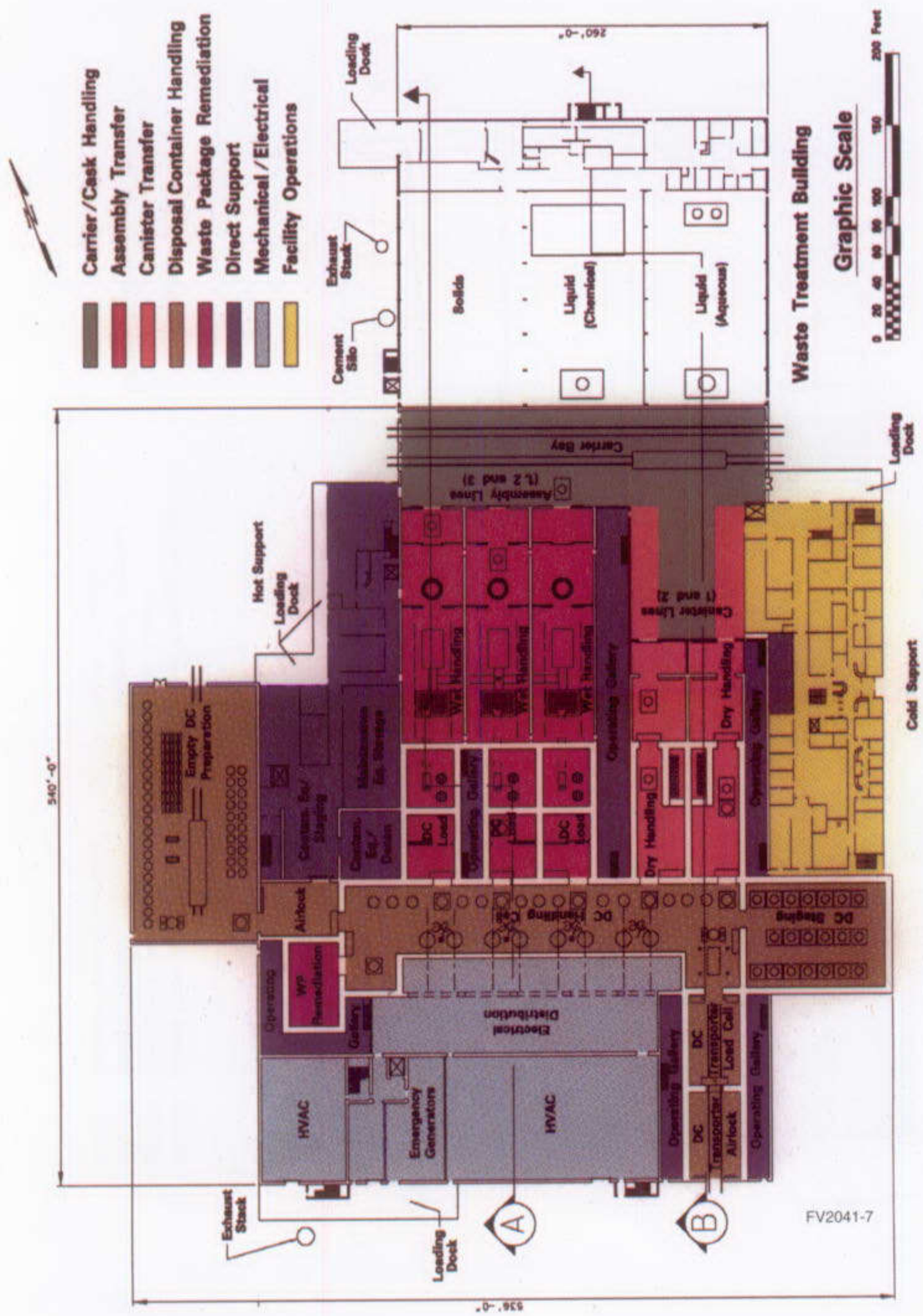


Figure 4-7. Waste Handling Building Floor Plan

system. Facility support areas will support the primary operations, including radiation protection, security, maintenance, and administration operations. Security and health physics areas will be required for facility surveillance, access control, radiological support, and health physics surveys of the facility and personnel. Office areas will be provided to support plant management, logistics, repair, engineering, records and document control, quality assurance, regulatory, and clerical activities. The following is a summary of the facility area (inside operating area) required for these operations:

Facility Functional Area	**Approximate Net Square Feet**
Waste Handling Systems Areas	146,000
Primary Support Areas	67,000
Facility Support Areas	54,000
Pool Support Area	9,000
Ventilation Equipment Areas	145,000
Miscellaneous Areas	19,000

Waste Handling Building Structure. One primary factor affecting the size of the Waste Handling Building structural elements will be the use of thick concrete walls as radiation shields. Based on the radiation levels expected from each waste form, the proximity of personnel to the radiation source, and the length of stay of affected personnel, some process areas may require concrete walls up to 5 ft thick.

The handling cells for cask opening, the canister and fuel assembly transfer cells, disposal container load and decontamination cells, and the disposal container welding and staging areas will require concrete walls about 5 ft thick to a height of 30 ft above the operating floor. Above 30 ft, the wall thickness can be approximately 3 ft. The roof structures will act as both shielding and tornado missile barriers. The roof structure will be supported by steel beams and concrete walls.

The disposal container handling areas will require concrete walls about 3 ft thick to a height of 10 ft, and a wall thickness of 1.5 ft above the 10-ft elevation. Secondary factors that will impact the building structure and foundation loads will be the heavy-duty overhead cranes, with handling capacities up to 140 tons, and 90- to 140-ton concentrated loads on the operating floor from casks and containers on transfer carts.

The pool area will consist of three water pools with up to six cask preparation pits. The water pools will require thick concrete walls lined with stainless steel plate. The stainless steel lining will provide primary containment. The concrete pool walls will support the steel lining and, together with the steel lining, form a part of the pool leak detection system.

The waste handling processes in the dry cask preparation areas, wet process pools, pits, equipment and tool rooms, and the carrier bay will require no facility radiation shielding, and can be constructed from metal clad structural steel framing. The carrier bay flooring will support loads from rail cars and trucks. Similarly, disposal container welder maintenance and service bays will have no radiation shielding requirements, and the superstructure can be structural steel framing with metal siding and roofing.

The facility cold support (non-waste handling) areas will include the administrative offices and laboratories, and have no radiation shielding requirements. The facility will be a steel-framed structure with sheet metal siding and will be a concrete slab on grade. The second floor concrete slab and metal deck roof will be supported by steel beams and columns. The disposal container preparation area will be an industrial type operations area without radiation shielding requirements, and will be constructed with light steel framing with sheet metal siding.

These areas will be separated from the facility primary structure to avoid interaction during an earthquake. The structural engineering effort for surface facilities was limited to preliminary proportioning of structural members (beams, columns, bearing walls, foundations, etc.) for estimated gravity loads such as structural member weights, equipment weights, and overhead crane lifted loads. Facility radiation shielding requirements resulted in wall thicknesses of 5 and 3 ft and floor and roof slabs up to 2 ft thick. Conservative structural member sizing was done in the preliminary analysis to account for forces and stresses due to earthquake loading.

Carrier/Cask Handling System. The carrier/cask handling system will be located in the Waste Handling Building (see Figure 4-8, Carrier/Cask Handling System). The system will receive rail and truck carriers containing loaded shipping casks from the carrier/cask transport system, unload the casks from the carrier, and transfer the casks to the wet assembly transfer system (for casks containing individual fuel assemblies or fuel assemblies in nondisposable canisters) or to the dry canister transfer system (for casks containing waste in disposable canisters). The system will also receive empty casks from the assembly transfer system and canister transfer system, and nondisposable canisters in overpacks from the assembly transfer system.

The system will load empty shipping casks onto carriers for transfer back to the Carrier Preparation Building, where they will be prepared for shipment back to waste generators. Empty transportation casks that require other than incidental maintenance will be loaded onto carriers for offsite shipment to the appropriate Regional Service Contractor. Nondisposable canisters in overpacks will be shipped to an offsite recycling or disposal facility. Four loading/unloading stations will be housed in the Waste Handling Building carrier bay. Two rail/truck lines will enter from one end of the bay. The carrier cask transport system will provide the prime movers, rail switching, and carrier ensemble turnarounds so that carriers can be moved in and out of the bay, and to minimize wait time. Waste handling, in the loading/unloading stations, will be provided by a carrier bay crane and two gantry-mounted manipulators. The support equipment will include cask lifting yokes, tooling, and fixtures required to support cask loading and unloading.

Assembly Transfer System. The assembly transfer system will be located in the Waste Handling Building (see Figures 4-9a and 4-9b, Assembly Transfer System). The system will receive shipping casks and cool the casks in preparation for handling in the pool. In the pool, spent fuel assemblies will be removed from the casks and nondisposable canisters will be cut open. The spent fuel assemblies will be transferred into baskets and stored or will be sent to the dryers, after which the assemblies will be loaded into disposal containers. Empty casks and canisters will be prepared for shipment offsite.

Cask unloading operations will begin with cask cavity sampling, venting, cool down, and lid unbolting. The cask preparation and pool transfer operations will be performed using an overhead crane. Casks containing individual fuel assemblies will be lowered into a pool, and the cask lid will be removed. Casks containing nondisposable canisters will have the lids removed at the preparation station; then the canister will be sampled, vented, and cooled before being loaded into a pool. Casks and canisters will be cooled down before lowering them into the pool to prevent steam formation in the pool and crud release from the outside surface of the fuel assemblies.

Assembly unloading operations for nondisposable canisters will begin by placing the canisters in an overpack and cutting the canisters open with a custom underwater lid severing tool. Fuel assembly transfer from open casks and canisters will then be performed using a wet assembly transfer machine. Empty casks and lids will be removed from the pool and returned to cask preparation to prepare them for reshipment. The nondisposable canisters will be removed from the pool and returned to cask preparation to prepare them for offsite disposal or recycling. In the pool, individual fuel assemblies will be placed in assembly baskets. The baskets will be staged in the pool or placed in a basket transfer cart when the assembly drying station in the assembly transfer cell will be ready to receive them.

The baskets, which hold four to nine assemblies, will be transferred up an inclined transfer canal out of the pool and into the shielded assembly transfer cell for drying. The operations will include moving the loaded baskets into the dryers and subsequent transfer of the dried assemblies from the baskets into disposal containers. Drying the assemblies prior to transfer into disposal containers drives off residual moisture which, in turn, will

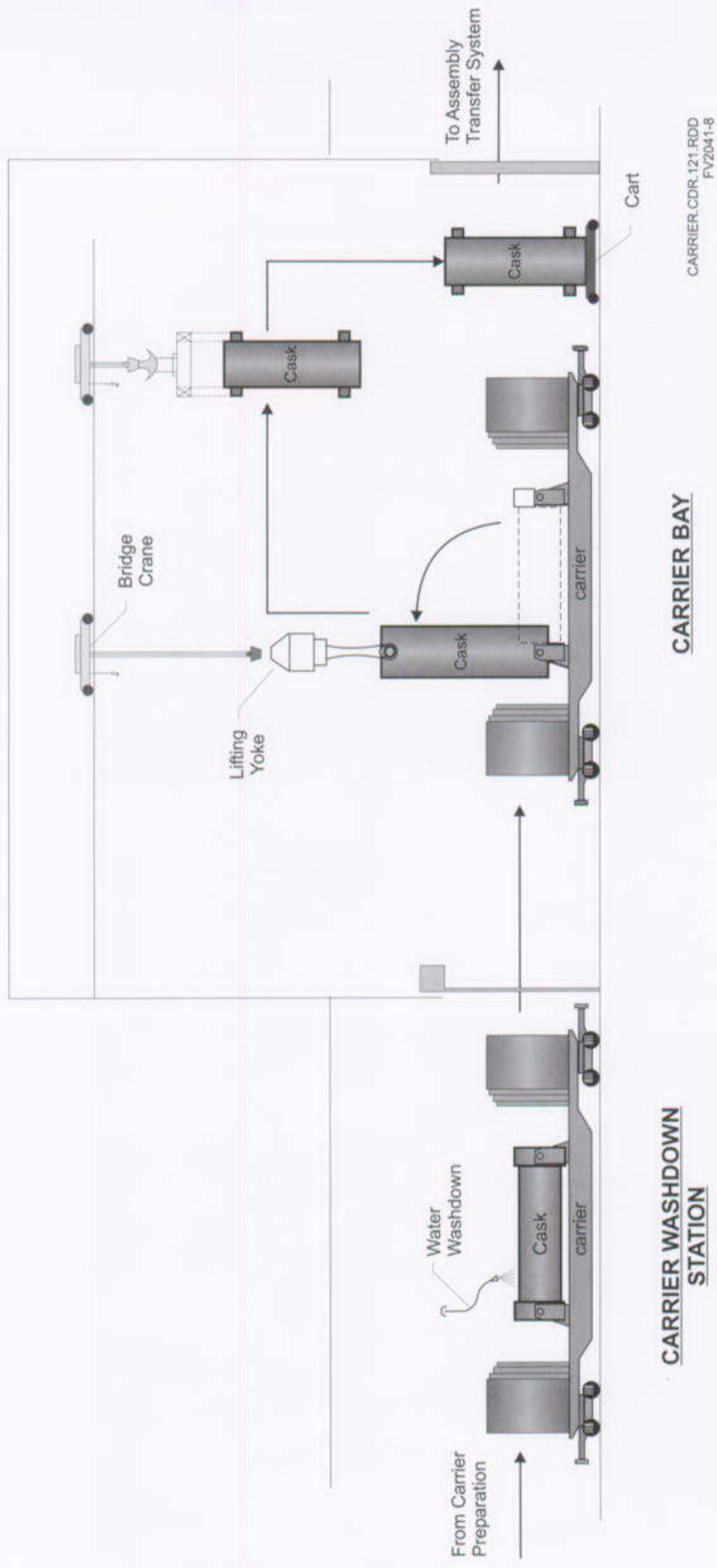


Figure 4-8. Carrier/Cask Handling System

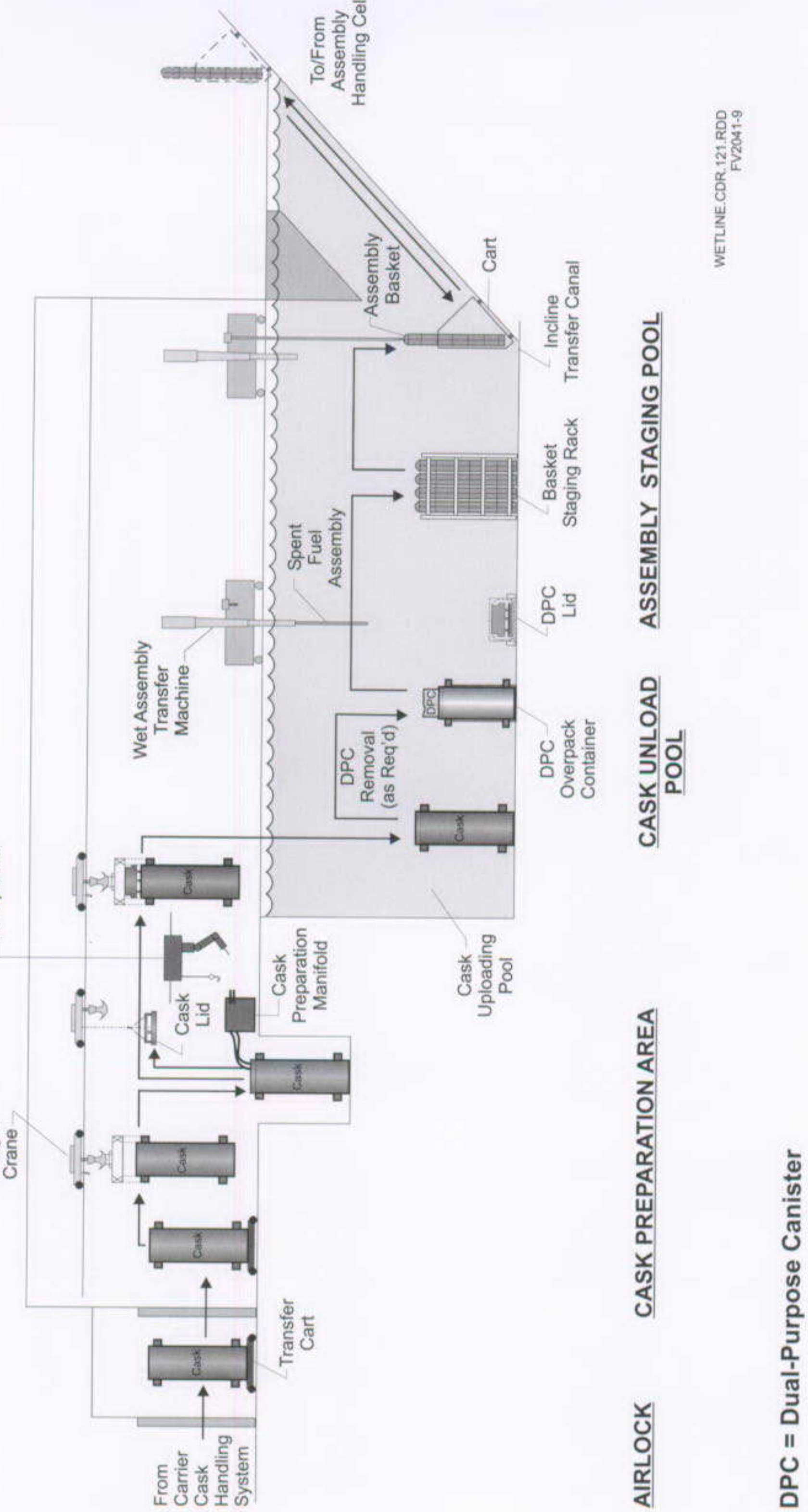


Figure 4-9a. Assembly Transfer System

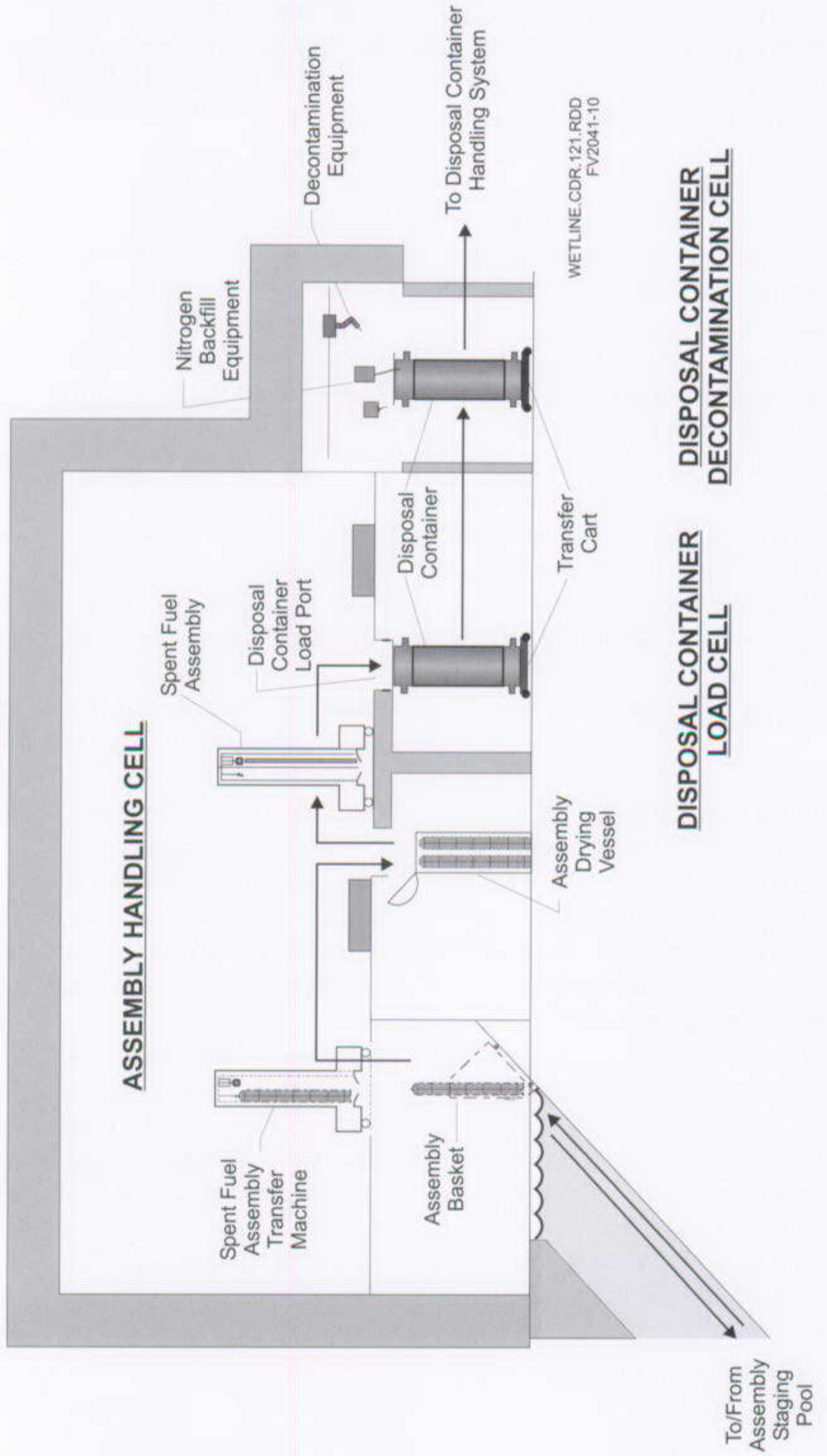


Figure 4-9b. Assembly Transfer System (Continued)

help to prolong the integrity of the waste form and the life of the waste package. Basket handling and disposal container loading operations will be performed with a dry basket and assembly transfer machine. After the assemblies are loaded into a disposal container, a temporary seal lid will be installed. The disposal container will then be decontaminated, filled with an inert gas, and prepared for welding.

Three identical assembly transfer lines will be provided. The lines can operate concurrently to handle the maximum number of casks received. Each cask unloading area will contain an airlock, one or more cask preparation pits, and a cask unloading pool. The handling equipment will include cask transfer carts, overhead bridge cranes, assembly handling machines, and nondisposable canister lid severing tools. The cask unloading pools will include assembly transfer canals for transferring assemblies between the three pools, wet staging racks for staging the assemblies, assembly transfer baskets for handling multiple assemblies, and basket transfer carts for transferring baskets from the pool to assembly drying units in the fuel assembly transfer cells.

The disposal container loading area will contain drying vessels, a disposal container loading cell, and a decontamination and inerting cell. The area will be supported by handling equipment consisting of a dry basket and assembly transfer machine, an overhead bridge crane, manipulators, and decontamination and inerting equipment. Various manipulators, lifting fixtures, carts, and tooling will be provided for basket, fuel assembly, and container handling.

Canister Transfer System. As described in the *Canister Transfer System Design Analysis*, the canister transfer system will receive shipping casks containing large and small disposable canisters, transfer the canisters from the casks into disposal containers, and prepare the empty casks for reshipment (CRWMS M&O 1997a).

Cask unloading will begin with cask inspection, sampling, and lid bolt removal operations (refer to Figure 4-10, Canister Transfer System). The casks will be transported into a canister transfer cell via a cart, where the cask lids will be removed, and the canisters will be unloaded. Small canisters will be loaded directly into a disposal container or will be stored until enough canisters are available to fill a container. The large canister will be loaded directly into a disposal container, and temporary lids will be installed. Shipping casks and related components will be decontaminated, as necessary, and empty casks will be prepared for reshipment. Two identical remotely operated and shielded canister transfer lines will be provided in the Waste Handling Building. The lines will be operated concurrently to handle canistered waste transfer throughput. Each cask preparation area will include an airlock, a cask preparation station, and a cask decontamination station.

Remote handling equipment will consist of cask transfer carts, manipulators, fixtures, and tooling required to perform cask unbolting, venting, lid removal, and decontamination. The canister handling areas will include a canister transfer station supported by remote handling equipment, including a bridge crane (sized to handle large canisters), a manipulator, and large/small canister lifting fixtures. A canister storage rack will be provided for the accumulation of small canisters to accommodate facility and equipment outages and for mixing canisters with various waste forms in a disposal container (referred to as co-disposal). A transfer corridor crane will traverse the canister and assembly handling cells for moving equipment in and out of the equipment maintenance bays.

Disposal Container Handling System. The system will prepare empty disposal containers for loading; receive full disposal containers from the assembly transfer system and canister transfer system; weld, test, and inspect the containers; and transfer them to the waste emplacement system, as discussed in Section 4.2. The system also will receive and handle retrieved waste packages from the subsurface and disposal containers with defective welds or other minor damage and route them to the waste package remediation system for corrective action (see Figure 4-11, Disposal Container Handling System).

Disposal container preparation will begin by unloading empty containers from the rail carrier,

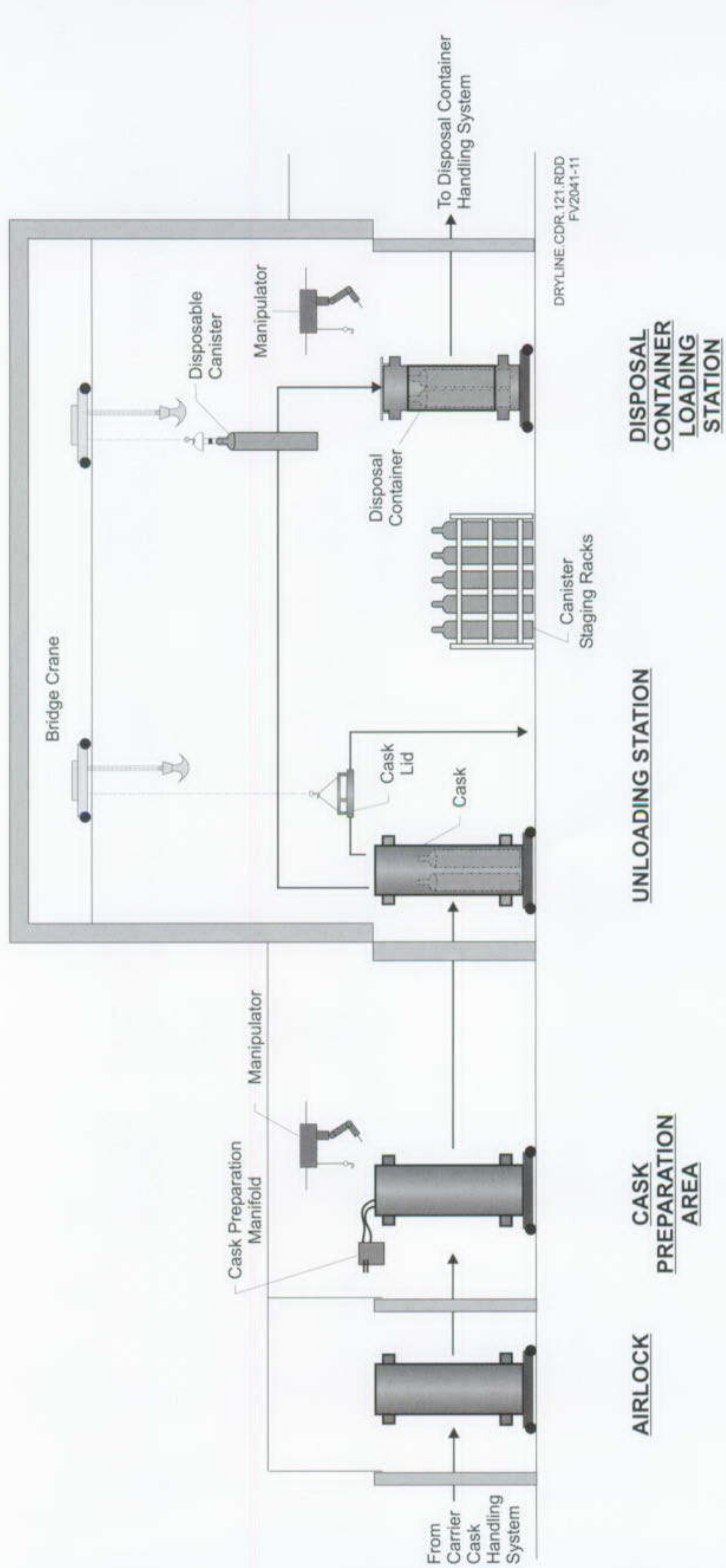


Figure 4-10. Canister Transfer System

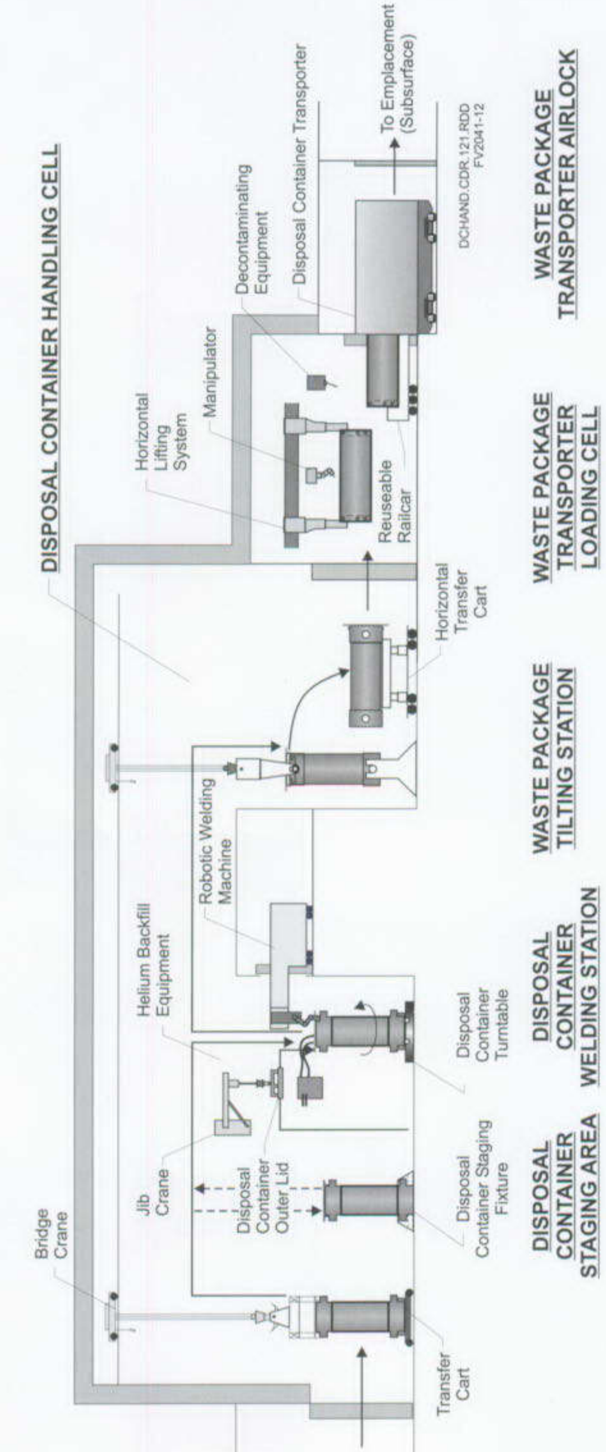


Figure 4-11. Disposal Container Handling System

staging the containers, installing handling collars and basket spacers, and transferring the containers to the disposal container handling cell. In the cell, empty containers will be transported to the assembly transfer system or canister transfer system for loading. Loaded containers will be transferred to the welding stations or to a staging area to wait for welding.

Welding operations will include positioning the containers on a turntable, removing temporary lid seals, welding the inner lid, inerting the container with helium, and installing and welding the outer lid. Each welding operation will be followed by nondestructive weld examination to determine if the container meets the criteria for placement in the repository. Upon meeting the criteria for placement in the repository, the unit will be termed a waste package. The transfer operations will include tilting the waste package to a horizontal position, removing the handling collars, decontaminating the waste package, and loading the waste package into a subsurface transporter.

The system will be located in the Waste Handling Building and will include the areas for empty container preparation, welding, waste package staging, tilting, decontamination, transporter loading, operating galleries, and equipment maintenance. These areas will operate concurrently to meet container throughput, which will be based on the waste emplacement schedules in Section 3.2.2. The empty container preparation area will be located in an unshielded structure and includes a carrier unloading dock. The loaded container handling equipment will be located in a shielded cell and will include a bridge crane, tilting station, container lid, and fixture preparation equipment. There will be eight disposal container welding stations, 20 staging (temporary storage) positions, five transfer carts connecting to the assembly transfer system and canister transfer system, and a tilting station. Welding operations will be supported by a remotely operated bridge crane and hoist, weld station jib cranes, and container turntables.

The waste packages will be prepared and loaded into an underground transporter within a shielded cell that will be supported by a remotely operated horizontal lifting system, decontamination equipment, manipulators, and a horizontal transfer cart. All handling operations will be supported by a number of fixtures, including waste package yokes, lifting collars, and lid collars. Remotely operated equipment will be designed to facilitate maintenance, and interchangeable components will be provided where appropriate.

Waste Package and Disposal Container Remediation System. The waste package remediation system will receive retrieved waste packages and defective or otherwise damaged disposal containers from the disposal container handling system and perform operations required for repair or examination of the containers. Repair or examination refers to either minor repair and examination or major operations requiring opening the container. The system will be located in the Waste Handling Building (refer to Figure 4-7, Waste Handling Building Floor Plan).

Waste packages will be retrieved for examination if failure or damage to the package has been detected. Examination may include remote visual and other nondestructive techniques, as well as analysis of physical samples required for performance confirmation. A general machining capability will provide limited repairs, such as welding defects in the package lids. Operations requiring the opening of a package will be infrequent, but will use the general machining capability to remove the inner and outer lids, as required. The system will interface with the disposal container handling system and will be designed to handle one container at a time.

A waste package or a disposal container will arrive at the cell on a transfer cart and will be placed at a repair/examination station by a crane. The package or container will exit the cell by the same method. Before opening, the container will be vented and the temperature, pressure, and gas composition internal to the package will be sampled. Inner and outer lids will be removed in a way that the package can be refurbished if possible. If a container is beyond repair, a temporary seal will be installed and it will be transferred by the disposal container handling system to the assembly transfer system or the canister transfer system for unloading. The empty container can be decontaminated and returned to the waste remediation system for further examination, or it can be shipped offsite for refurbishment or disposal.

The system will include a wide variety of remotely operated components, including a crane with lifting yokes and fixtures; viewing systems and manipulators; machine tools for opening and/or repairing containers; analyzers and test instruments; and equipment used to take samples, perform nondestructive examinations, and collect data.

Waste Handling Building Ventilation System. The Waste Handling Building ventilation system will supply fresh air and control the environmental conditions in the facility operating, equipment, and support areas. The system will operate in conjunction with the facility physical barriers to control the air flow and pressure in each controlled area. The exhaust air will be filtered to prevent the spread of radioactive contamination, exposure to personnel, and releases to the environment. Airborne radioactive release rates will be controlled within environmental standard limits, and releases to the personnel occupancy zones will be controlled to below health safety standard limits. The primary ventilation zone will be enclosed by the facility physical protection barriers (shield walls and doors) that confine the primary waste handling equipment and waste forms. The secondary zone will have a potential for contamination, and the tertiary zone will normally be free of radiation or contamination, as discussed in the *Surface Nuclear Facilities HVAC Analysis* (CRWMS M&O 1997aj). Refer to Figure 4-12, Environmental Design Conditions.

The ventilation system will control the three confinement zones, establish the airflow, and controlling the facility and area room pressures. Air flows from the outside environment to operational areas and then is discharged to the filtration system and the exhaust stack (refer to Figure 4-13, Waste Handling Building - Ventilation Flow Diagram).

The system will maintain the pressure differentials between the primary zone (low pressure relative to the secondary zone), secondary zone (negative pressure relative to the outside environment), and tertiary zone (low pressure relative to the outside environment, but higher than the secondary zone). Each zone will consist of an independent ventilation system with separate instruments and controls, and will be integrated (coordinated) by a central ventilation control system. The supply air segment of each ventilation system will consist of a tornado-missile-resistant outside air intake, capable of withstanding the impact of a missile such as an airborne piece of wood or metal carried by a tornado; an air handling unit with filters, heating and cooling units, and sprayers; and supply air fans that will distribute the air through supply air ducts. Dampers will be controlled to maintain the air conditions to each zone area. The exhaust air segments of the systems will provide contamination confinement. Each zone system will consist of exhaust air ducts, high efficiency particulate air filter units, exhaust air fans, and tornado dampers. The three ventilation systems will exhaust to a single facility exhaust air stack which will contain radiation monitoring equipment.

The emergency ventilation system will be powered by an emergency power system located in the facility, as discussed in the paragraph on the Waste Handling Building electrical system.

Radiation Access Zones in the Waste Handling Building. Various areas in the Waste Handling Building will be designated to have radiation levels that either preclude human occupancy or in which occupancy will be controlled. These areas will be designated as radiation access zones, which will be defined as areas with radiation levels that potentially fall within boundaries that correlate with the limits in 10 CFR 20 and with the more conservative current operational policies that are in practice at the Nevada Test Site and other DOE sites, as described in DOE radiological control manuals. The object of the radiation access zones will be to provide a design framework that realistically limits radiation exposures to as low as reasonably achievable levels given the state of technology, economics, and the benefits to the public health and safety. The radiation access zones that will be designated in the Waste Handling Building for the VA reference design are described in Table 4-1, Waste Handling Building Radiation Access Zone

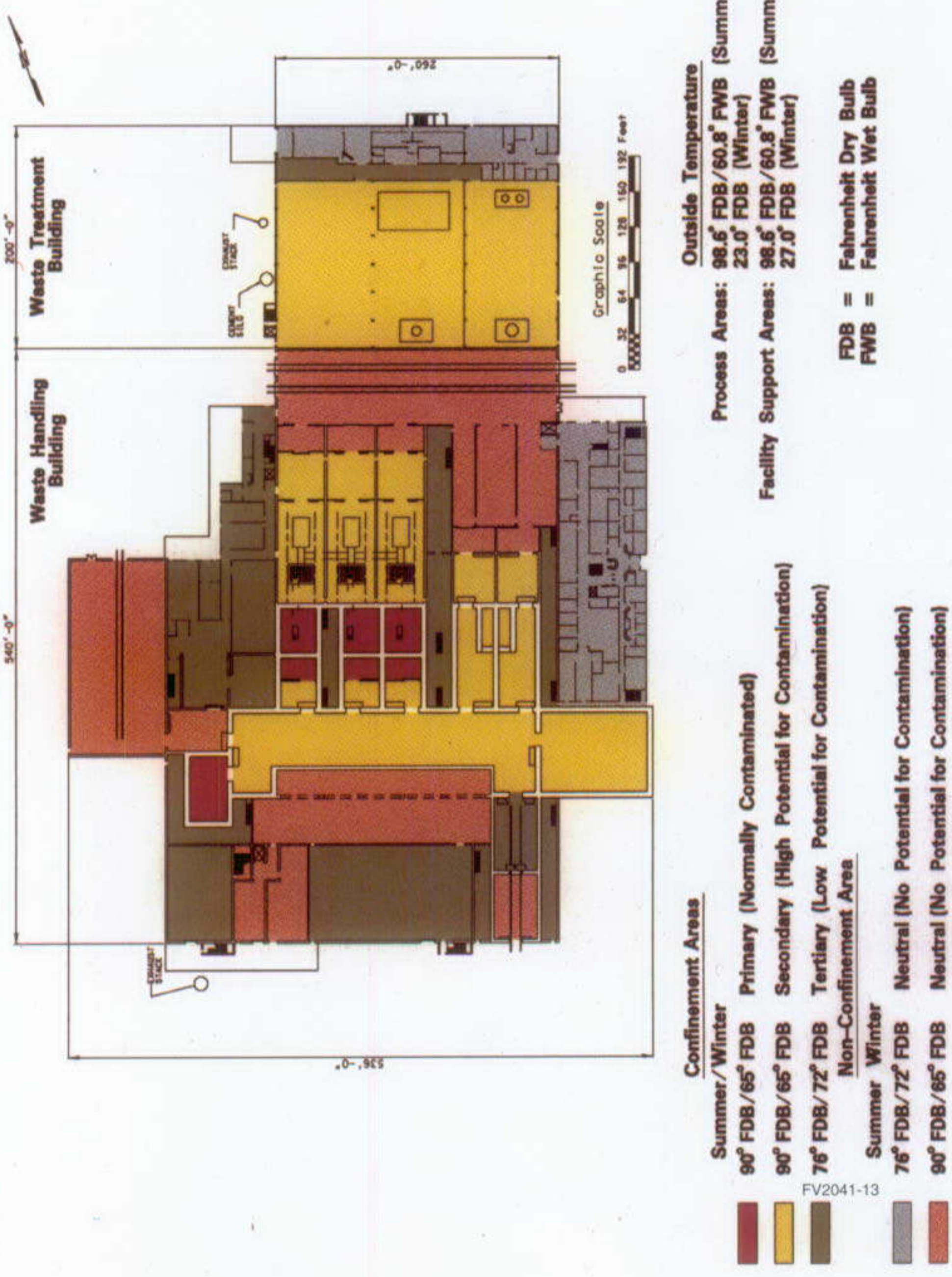
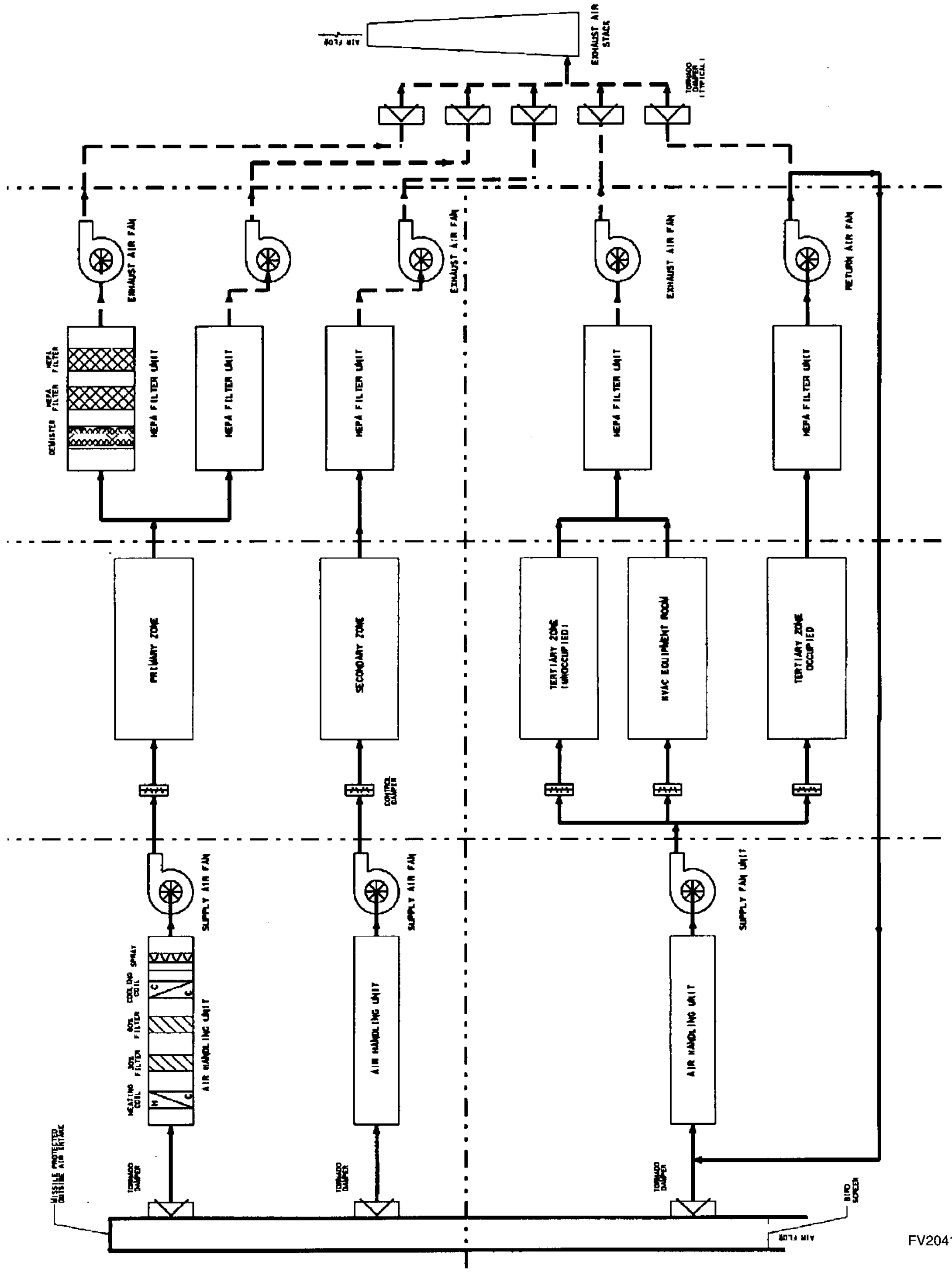


Figure 4-12. Environmental Design Conditions



Legend
HEPA High efficiency particulate air
HVAC Heating: ventilating: and air conditioning

Figure 4-13. Waste Handling Building-Ventilation Flow Diagram

Table 4-1. Waste Handling Building Radiation Access Zone Designations

Radiation Zone	Description
Very High Radiation Area >500 rem/hour	This zone occurs where sources (e.g., spent fuel assemblies, high-level radioactive waste canisters) are transferred unshielded to either storage or disposal containers. **No Normal Access Permitted.**
High Radiation Areas >0.1 rem/hour to <500 rem/hour	This zone covers the loaded disposal container handling areas and cask operations areas. **No Normal Access Permitted. Access permitted under controlled conditions with Special Radiation Work Permits.**
Radiation Work Permit Areas >0.25 mrem/hour to <0.1 rem/hour	General Work Areas, Carrier Bay, Cask Preparation Area
Unrestricted Operations Area <0.25 mrem/hour	General areas of the operating corridors

Designations. Figure 4-14, Radiation Levels, depicts the maximum dose levels anticipated during the handling of waste in the Waste Handling Building.

Waste Handling Building Radiological Monitoring System. Area radiation, continuous air, and criticality monitors will be installed in the operating and equipment areas, including facility ventilation filtration and exhaust stack equipment, waste handling pools, and dry waste handling areas. The monitoring system will monitor trends in any increases of radiation from expected area radiation levels, and will provide warnings and advisories before unsafe levels will be reached. The detection instruments will perform self-tests on operating status and calibration, record the results, and report anomalies and failures. The system will operate in coordination with the health safety system, which will maintain personnel dosimeters as well as portable and full body radiation monitors to personnel who may be exposed during access to the facility and controlled areas. Essential radiation data and alarms will be provided to facility personnel, security, health physics, and emergency response systems.

Waste Handling Building Electrical System. The Waste Handling Building electrical system will perform the function of conditioning, distributing, controlling, and monitoring power to waste handling facility users. The system will consist of the step down (voltage reduction) transformers, switchgear, controls, uninterruptible power supplies, and electrical distribution subsystems required to power facility material handling, facility lighting, ventilation, instrumentation and controls, radiation monitoring, and process equipment. The process equipment will be associated with facility waste collection, vacuum, industrial air, gas supply, decontamination, and pool conditioning.

Facility power will be provided from 12.47-kVA electrical switchgear circuits A and B at the site power switchgear (refer to Figure 4-15, Normal and Emergency Power System). Power required for Waste Handling Building safety class equipment and nonsafety class equipment will be distributed throughout the facility from electrical buss A and B. The power will be distributed to facility electrical loads through 480-volt panel boards, motor control centers, and load centers. Safety class equipment will include the ventilation required to maintain radiological confinement, as well as selected cask and waste handling equipment, security, communications, and essential health and safety systems.

Safety Class equipment will be required for mitigating and confining a radiological release, the most important being the facility ventilation exhaust fans. The fans will maintain facility air flows and pressures to confine an unlikely radiation release to uninhabited areas and filter the emissions that might be released to the environment. Safety Class operating equipment loads will also be provided power from a 1,000-kVA, quick-start emergency diesel generator located in the facility. The emergency generator will automatically start on loss of utility power and receive fuel from a dedicated diesel fuel tank sized to handle safety loads for 24 hours without refilling. The generator will also be Safety Class equipment.

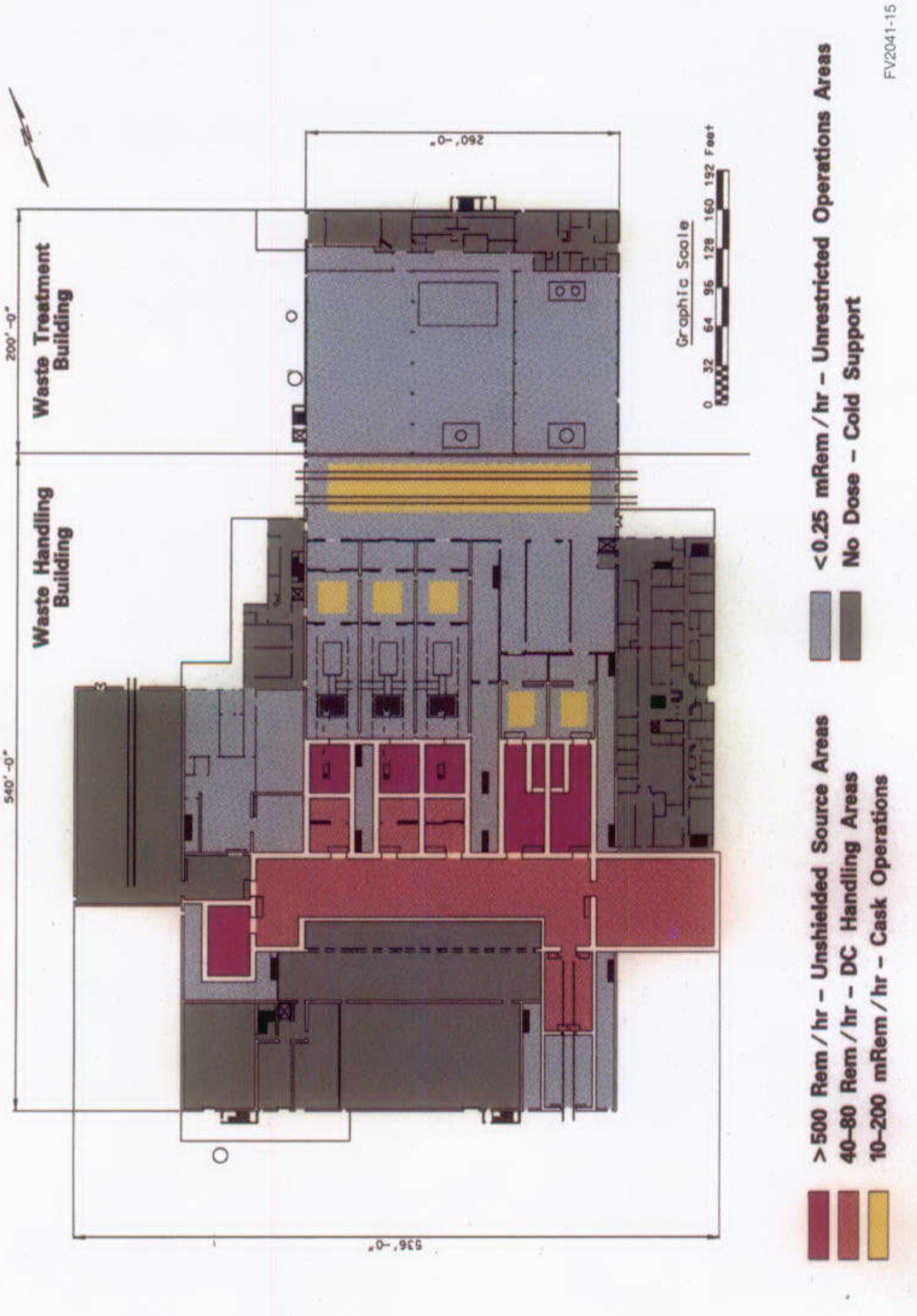


Figure 4-14. Radiation Levels

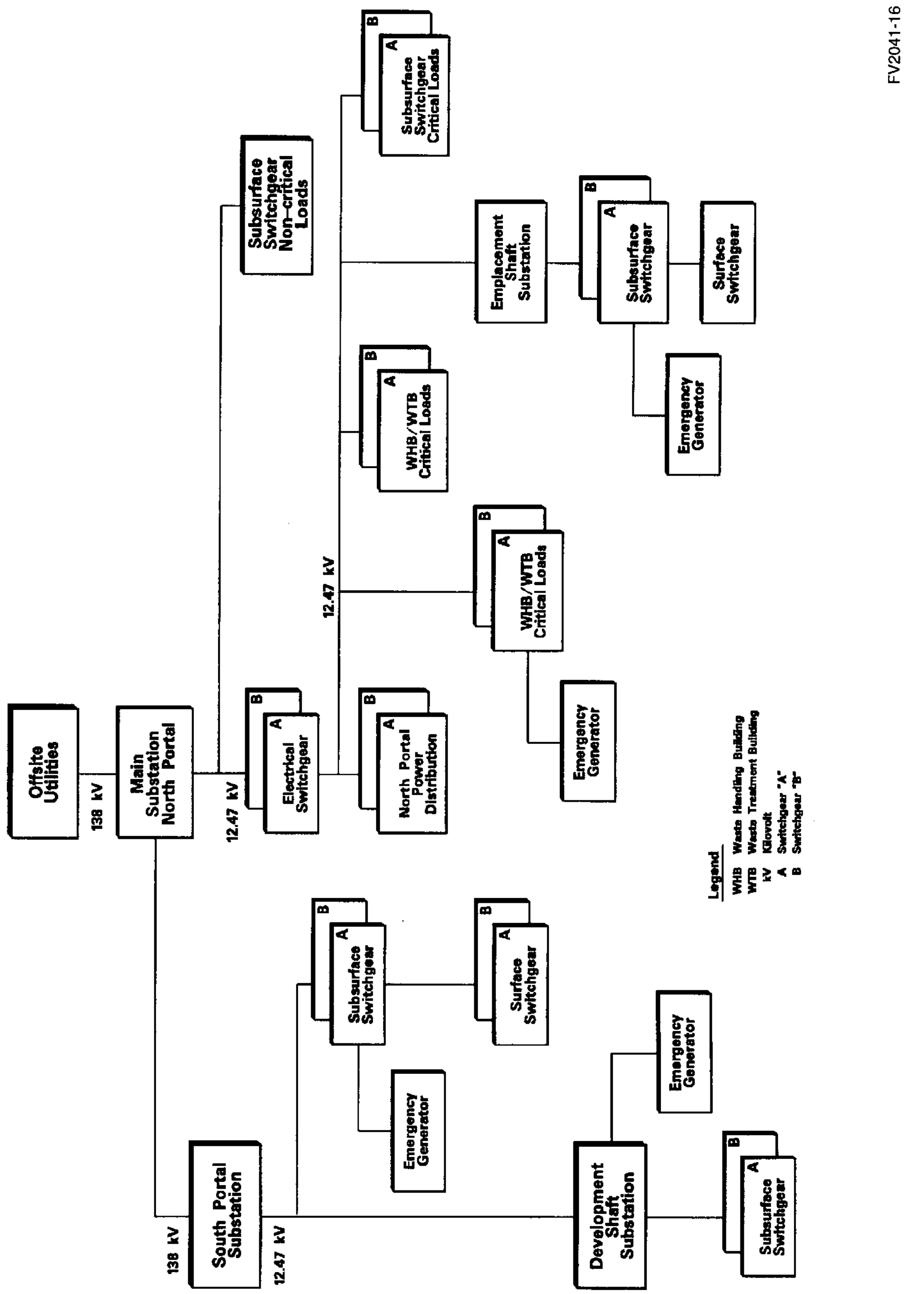


Figure 4-15. Normal and Emergency Power System

Facility safety-related equipment, including facility instrumentation, radiation monitoring, and computer systems, which cannot tolerate a power interruption, such as a temporary power loss or brown out (low voltage), will be supplied with uninterruptible power supplies. These supplies will consist of storage batteries, battery chargers, and converters that convert direct current voltage to alternating current voltage. On utility power failure, the safety backup power will be transferred instantaneously (bumpless) from the storage battery equipment, or can be switched to emergency power.

Waste Handling Building Fire Protection System. The Waste Handling Building fire protection system will perform the function of detecting and automatically suppressing fire in the facility. The fire detection subsystem will automatically monitor and sound an alarm for fire and potential fire conditions based on smoke, heat, and rate of temperature rise conditions. The fire detection system will automatically alert facility personnel, the fire house, site security, and other emergency response stations of the fire conditions, and initiate the facility fire suppression subsystem in the affected areas.

Smoke and heat detection, fire pull boxes, and alarm instrumentation will be installed throughout the facility in accordance with national and local fire codes and radiological facility standards. The suppression system will automatically initiate wet sprinkler or water deluge equipment when fire or smoke is detected. The system will include the required dry chemical extinguishers, fire hoses, and pull box stations located throughout the facility.

The fire protection system will provide timely and reliable initiation of fire alarm and suppression equipment and accurate location of the fire to emergency response and support personnel. The system will operate in conjunction with the facility ventilation system to detect smoke and fire in specific areas and to minimize the conditions through controlled ventilation. The detection and alarm system will be powered from an uninterruptible power supply.

The system will be installed in all facility operating and equipment areas, and will include posted instructions for the safe and controlled egress of facility personnel. The suppression systems will be connected to the site fire water system. In radiation controlled areas and areas where there will be a potential for contamination, sprinkler water from the automatic fire suppression system will be collected by floor drains and routed to a holding tank. If contaminated, it will be treated before disposal. Wet fire suppression systems will not be used in areas with open access to dry spent nuclear fuel.

4.1.4.3 Waste Treatment Building

The Waste Treatment Building will house the site-generated radioactive waste handling system which collects and prepares the site-generated low-level radioactive solid, liquid, and mixed waste for disposal. Waste will be referred to in this section as low-level radioactive waste in either solid or liquid form. The system will control the collection of waste and treats it prior to packaging for disposal offsite. The *Waste Handling Operations - Dose Assessment* concluded that the radioactivity of the waste will be low enough that no special facility features will be required to meet NRC radiological safety requirements for shielding and criticality (CRWMS M&O 1997an).

The building will be adjacent to the Waste Handling Building carrier bay. The facility will house the handling equipment, process tanks, piping, instrumentation, offices, and personnel involved in the collection and processing of chemical liquid and solid waste from the Waste Handling Building preparation and handling processes. The system will also contain equipment, tanks, and piping for dewatering of spent resin that has been used for purification of the three pools in the Waste Handling Building (refer to Figure 4-16, Waste Treatment Building Floor Plan).

The Waste Treatment Building will be a two-story, high-bay industrial, steel frame structure. The main operating floor will be a cement slab on grade. The superstructure will be braced-frame structural steel, with metal siding and a metal deck roof. Support personnel offices will be located on the ground floor. An elevated floor or mezzanine

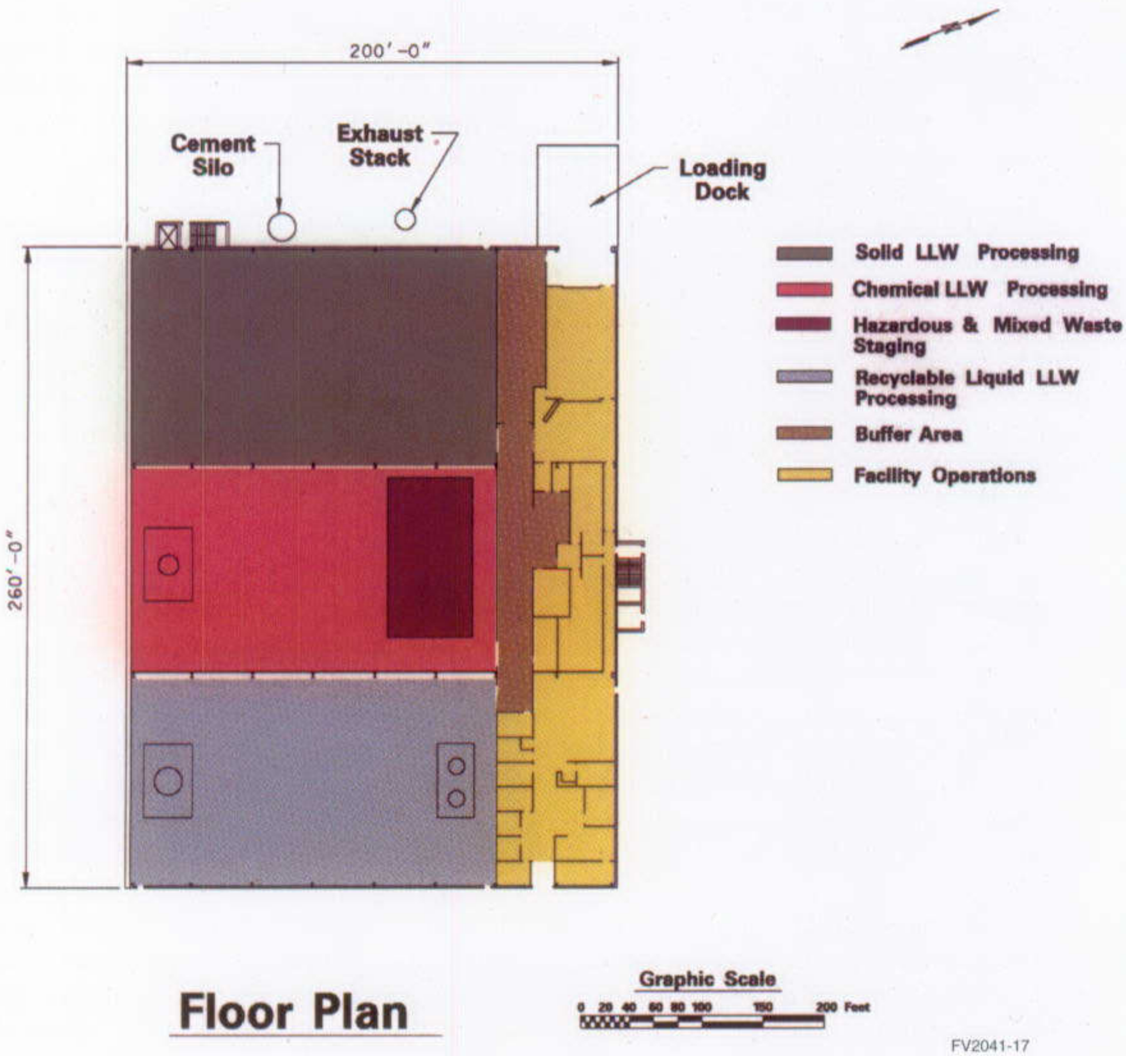


Figure 4-16. Waste Treatment Building Floor Plan

will be located above the personnel offices for the facility mechanical equipment.

Site-Generated Radiological Waste Handling System. The site-generated radiological waste system will receive radioactive liquid and solid low-level radioactive waste generated at the waste handling facilities within the controlled area and will safely process the waste, as discussed in the *Secondary Waste Treatment Analysis* (CRWMS M&O 1997ae). The process will include separating solid and liquid waste, processing liquid waste to enable reuse of water, placing the waste in containers so that it will be suitable for disposal, and storing waste to wait for offsite disposal. The Waste Treatment Building will house the systems that process the waste streams described in this section.

In the surface waste preparation and handling operations, solid and liquid low-level radioactive waste will be produced, most of which will be from the Waste Handling Building. The operations will include decontamination of shipping casks, material from a facility transfer cell, and radioactive waste handling equipment. Some of these operations will be performed with chemicals and wipes, which will also become waste. Waste will also be collected from the Waste Handling Building pool skimming and filtration equipment; cuttings from the process of opening nondisposable canisters; contaminated tooling and clothing; facility ventilation filters; chemical sumps; and carrier and transporter washdowns.

Used (opened and unloaded) dual-purpose canisters will be considered low-level radioactive waste; they will be placed in an overpack suitable for shipping. The used canisters will be packaged for offsite shipment at the Waste Handling Building and will not be processed by the system described in this section.

The majority of the Waste Handling Building liquid waste will be pumped through piping to the Waste Treatment Building process systems. Other waste will arrive in sealed containers on the site transportation system. Recyclable liquid waste will be treated and made available for users. Chemical liquid and solid waste that will not be recyclable will be grouted (immobilized in cement) and packaged for disposal. Compactible solid waste will be sorted, compacted, and readied for disposition. Noncompactible solid waste will be cut into pieces, or otherwise physically reduced in size, and will then be grouted and packaged. All nonrecyclable waste will be processed and compacted in 55- or 85-gallon drums and readied for shipment to a licensed low-level radioactive waste disposal facility (refer to Figure 4-17, Recyclable Liquid Low-Level Radioactive Waste Treatment, Figure 4-18, Chemical Liquid Low-Level Radioactive Waste Treatment, and Figure 4-19, Solid Low-Level Radioactive Waste Packaging).

Mixed waste will be waste material exhibiting the characteristics of both low-level radioactive waste and hazardous waste. It is not anticipated that this waste will be produced as a result of normal operations, and the system will eliminate or control the generation of mixed waste. An allowance will be made for staging a small quantity of this waste within the Waste Handling Building. The mixed waste will be collected, packaged, and temporarily staged, pending offsite shipment.

Waste Treatment Building Ventilation System. The Waste Treatment Building ventilation system will supply air and control the operating zone pressure and environmental conditions to equipment and personnel areas within the facility. The system will maintain the minimum required pressure differentials in the low-level radioactive waste process and handling areas to facilitate controlled air flow, and control the temperature inside the facility to be within prescribed equipment and occupancy safety limits. As discussed in the *Surface Nuclear Facilities HVAC Analysis*, the ventilation system design and operation will be similar to the Waste Handling Building system previously described; however, there will not be a primary ventilation zone, the equipment will not be safety class, and emergency backup power will not be provided (CRWMS M&O 1997aj).

Airborne contamination will be removed and airflow will be controlled away from penetration barriers to protect personnel from radiation exposure and minimize inadvertent release of radioactive particles to the site boundary. Fire protection,

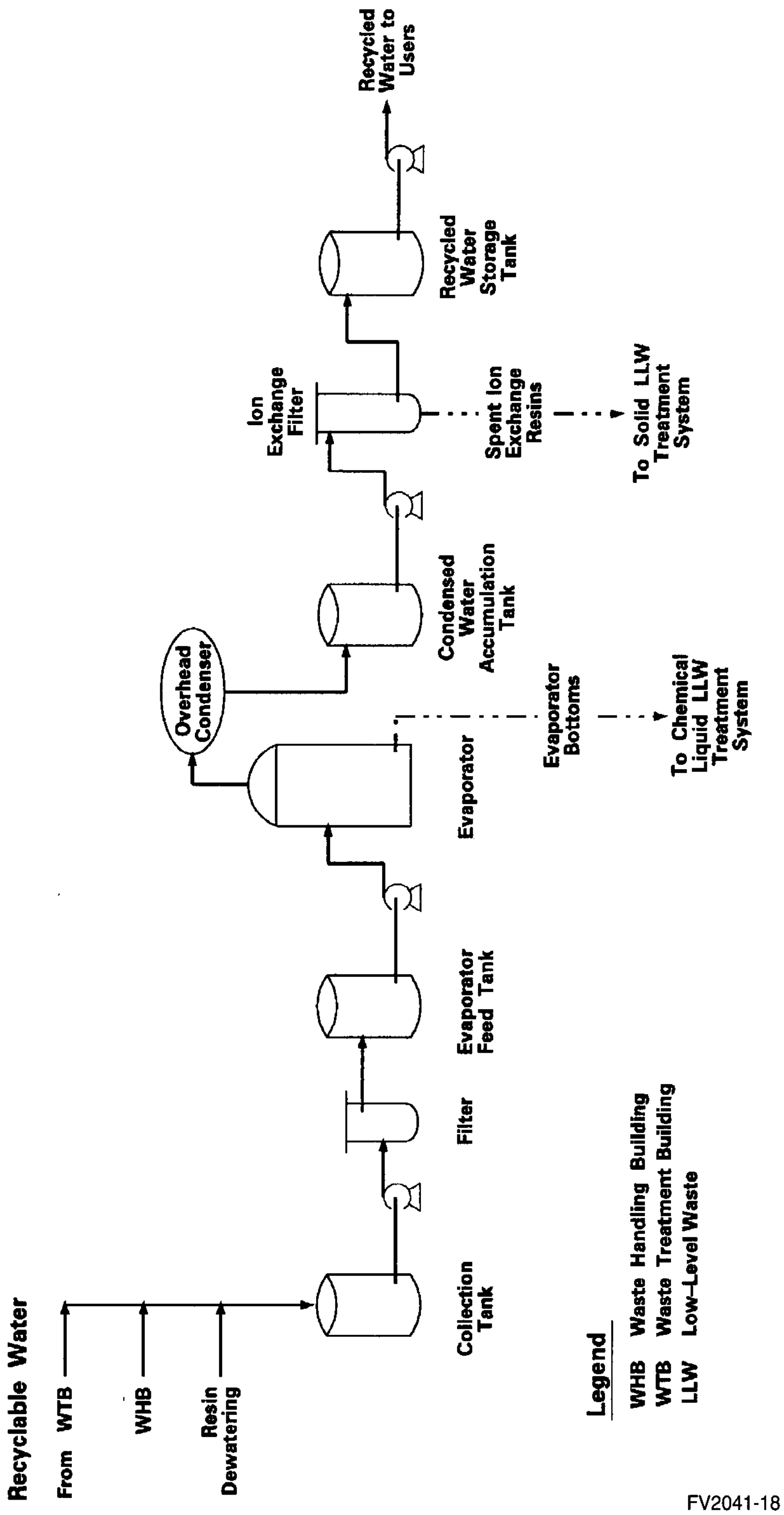


Figure 4-17. Recyclable Liquid Low-Level Radioactive Waste Treatment

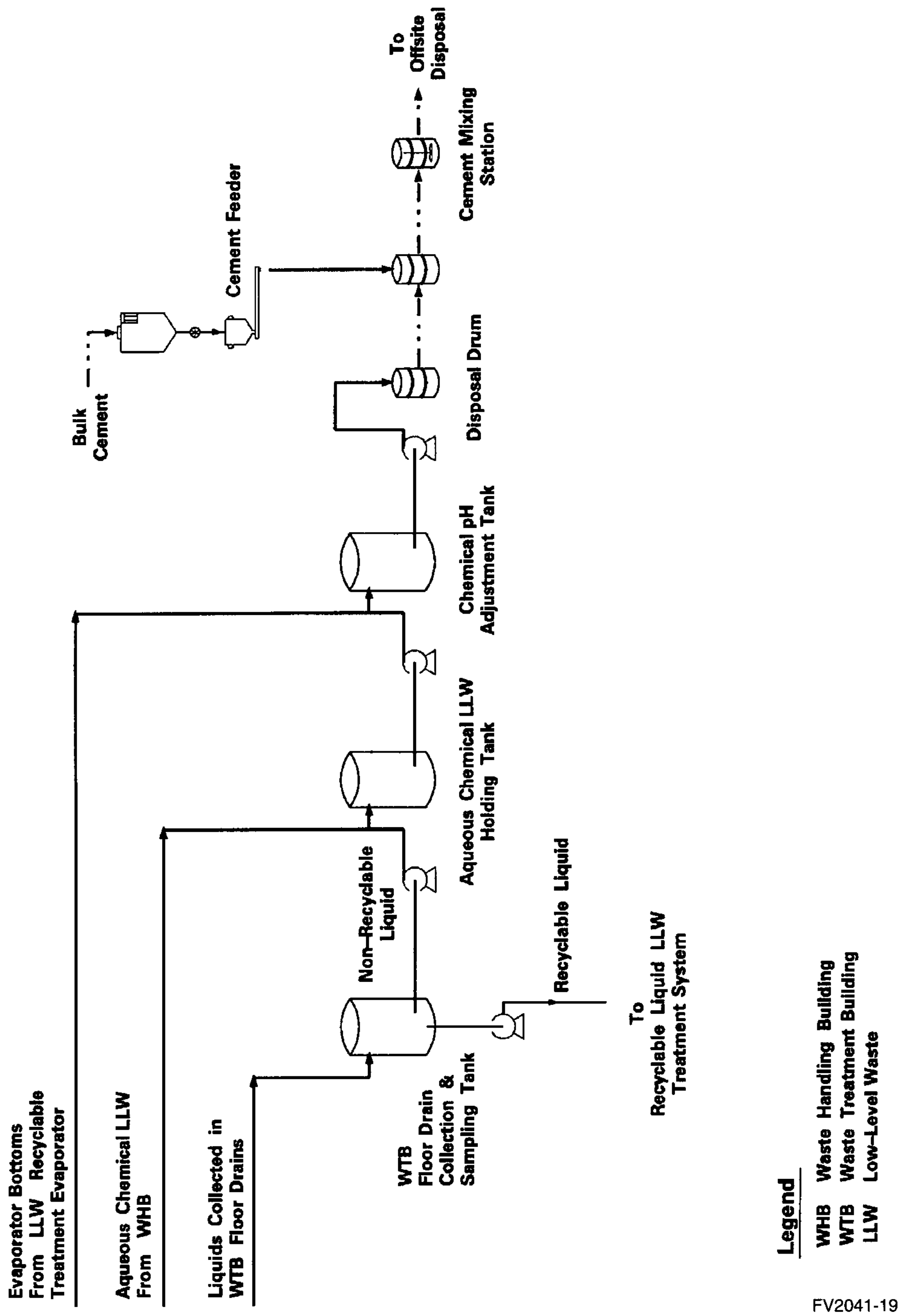


Figure 4-18. Chemical Liquid Low-Level Radioactive Waste Treatment

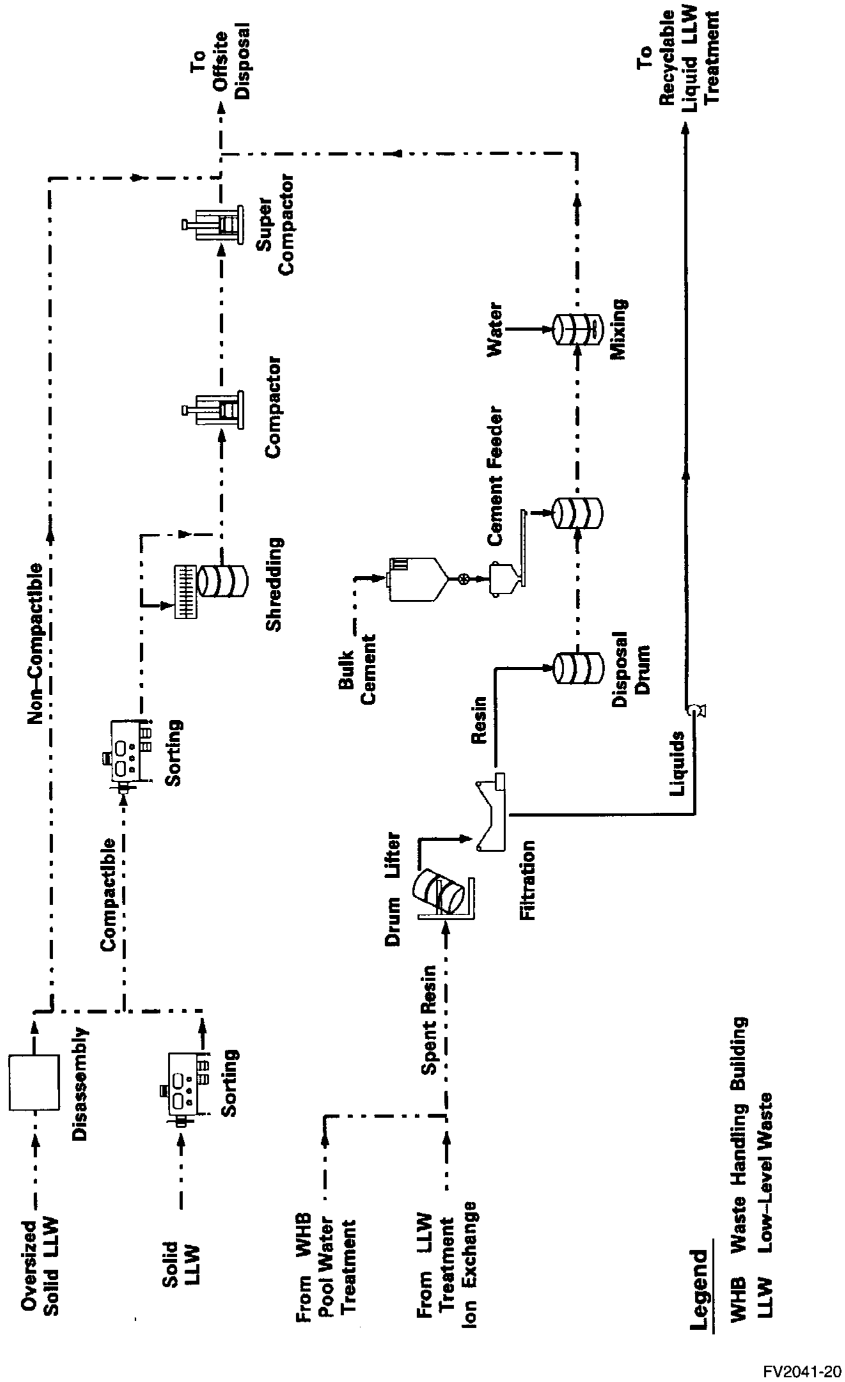


Figure 4-19. Solid Low-Level Radioactive Waste Packaging

radiation monitoring, and leak detection systems will also be included in the design.

4.1.5 North Portal Balance of Plant Facilities

The balance of plant area will include the facilities required for the management and support of waste handling, repository development, and waste emplacement operations. There will not be radiological handling activities associated with balance of plant facilities, which will be located outside the controlled area (refer to Figure 4-1 for the general location of the balance of plant facilities at the repository). The majority of the facilities will be of structural steel frame construction with insulated roofs and siding. The computer center will be constructed of reinforced concrete for security purposes. The facilities will be constructed in accordance with the Uniform Building Code (ICBO 1997).

Administration Building. The administration building will include the site operations management and planning, waste transportation coordination, and administration offices. Also included will be areas and offices for training, performance confirmation laboratories, a computer center, and a cafeteria. The transportation coordination center will maintain the offices and transportation dispatcher communication systems. The computer center will be in a hardened portion of the facility and will house the central computers, network servers, workstations, and computer peripheral equipment required for site planning and management, dispatch, monitoring and control, office, and data systems.

The training center will provide the classrooms, training equipment, offices, and records centers required to train personnel and plan and maintain the site personnel training and certification program. Laboratory space will also be provided for collection and analysis of the performance confirmation program data and samples.

Fire Station. The fire station will be a major element of the onsite emergency response system. Fire station personnel will respond to fires and accidents to provide required treatment and to stabilize accident conditions. The station will maintain fire and rescue vehicles, equipment, and trained professionals required to respond to fires and other onsite emergencies.

Medical Center. The medical center will be located adjacent to the fire station. It will maintain a full time doctor, nurse, treatment rooms, and the medical stores required for treating injuries and illnesses. A communications area will coordinate with offsite professional medical, fire, rescue, mining, and radiological emergency organizations.

Central Warehouse. The warehouse will maintain an inventory of critical equipment spares, materials, and consumables required for maintaining operations.

Central Shops and Motor Pool Facility. The shops will provide the basic machining, carpentry, plumbing, mechanical, and electrical repair capability to support general site and utility repair and improvements. The motor pool will provide for general conventional vehicle maintenance and fueling.

Mockup Building. The Mockup Building will house the critical systems required to test the operational material handling systems, develop material handling procedures, support development of new handling equipment and upgrades to existing equipment, and train personnel. Wet and dry waste handling test beds will be available in the Mockup Building.

Utility Building. The Utility Building will be located next to the cooling tower, and consist of a mechanical equipment room, an office, and restrooms. The equipment room will house facility heating, cooling water, water treatment, and industrial air systems. The cooling water system will include chillers, pumps, and tanks to supply chilled water to North Portal facility ventilation systems. The water treatment system will treat make-up water (water lost mostly due to evaporation) for facility systems, with the major user being the cooling tower. The air system will provide industrial quality air for distribution.

4.1.6 Security and Safeguards System

The security and safeguards system will consist of facilities, subsystems, and equipment throughout the site. The system will perform the surveillance and safeguard functions required to protect the repository from unauthorized intrusion, radiological sabotage, theft, and the diversion of nuclear material. The system will include site security barriers, gates, automated surveillance, badging of personnel and visitors, and record subsystems required to monitor and control access to all site areas and facilities. To prevent unauthorized access and theft and to provide timely detection of contraband, including explosives, arms, and hazardous or dangerous substances, security inspections will be performed at three access points to the surface facilities. Other security inspections will be provided at the construction and emplacement areas associated with the subsurface operations.

4.1.7 Health Safety System

The health safety system will monitor personnel exposure to hazardous substances and radiation. The system will monitor operational areas for hazardous materials and personnel for exposure to radiation and hazardous substances, including hazardous operating and maintenance conditions. The system will maintain a safety program and a safety training program, perform health and safety surveillance and surveys, develop and maintain safety procedures, maintain health records, and manage corrective action for unsafe conditions. The safety system will also provide breathing air for emergency and off-normal waste handling operations and control access to radiologically controlled areas based on procedural restrictions, including personnel radiation exposure histories.

The health safety system will monitor access to the site radiological facilities and controlled areas. Sufficient health safety coverage will be provided for all radiological and hazardous areas to ensure that workers will be protected from exposure to hazards, and that the safety measures, safety or protective clothing, and equipment decontamination facilities will be available and in useable condition. Personnel health safety will be monitored, tracked, and recorded, and records will be used to control area access.

The system will operate in conjunction with the administration system and the medical facility to collect and maintain personnel health data. Health physics laboratories, offices, and calibration shops will be located in the waste handling building in the controlled area. These shops will maintain the radiological and hazardous detection equipment and personnel dosimeters and perform personnel surveys during operations.

4.1.8 Surface Environmental Monitoring System

The surface environmental monitoring system will monitor the surface areas and groundwater for radioactive and hazardous substance release into the environment. The system will monitor and collect environmental data from key site areas for airborne particles containing radioactive or hazardous components. It will also monitor facility effluents and tests wells for radiation and hazardous material, gather meteorological and seismic data, and collect soil and vegetation samples for analyses.

The subsystems of the environmental monitoring system will be as follows:

- Air quality monitoring system
- Liquid effluent monitoring system
- Meteorological monitoring system
- Surface environmental monitoring system
- Groundwater monitoring system
- Seismic monitoring system

The system will monitor trends in environmental data, record the data, alarm the data thresholds at the central stations, and alert operations authorities when established thresholds will be exceeded. The system will include sensors, instrumentation, analyzers, and the manual and automated data collection equipment required to collect, process, display, store, and archive site environmental data and provide periodic and event reports.

4.1.9 Emergency Response System

The emergency response system will respond to accident conditions at or near the repository and return or stabilize the conditions to as close to normal as possible. The system will maintain the emergency and rescue equipment, communications, facilities, and trained professionals required to respond to fire, radiological, mining, industrial, and general surface and subsurface accident events.

The system will control rescue and site evacuation services and provide medical care to personnel. It will coordinate this capability with offsite organizations as required to mitigate accident conditions and treat the injured. The primary emergency response subsystems will consist of the fire station, medical center, and emergency response center.

The emergency response system will be located in the Administration Building, in the balance of plant area. It will be supported by the health safety and underground rescue capabilities at the site and will maintain special rescue equipment stores. Emergency response teams will include onsite and offsite emergency specialists and professional people, augmented by special teams of highly trained onsite personnel. The site systems will be operated full-time and will be supported by offsite professional medical, fire, and rescue organizations. Communication with these organizations will be accomplished through a redundant emergency communications capability consisting of a land line and microwave systems. Emergency facilities, systems, and personnel will be maintained at a high degree of readiness. The system will interface with all site operational areas and facilities and maintain a full-time interface with the site and offsite fire, medical, and transportation systems.

4.2 REPOSITORY SUBSURFACE FACILITIES

This section describes the design and layout of the repository subsurface features, the ground control system that will support excavated openings, the waste transportation and emplacement system components, the subsurface ventilation systems, the monitoring and control systems, and the utilities (water) required to support construction and operation of the repository. In addition, it describes the equipment and processes to be used to retrieve any emplaced waste packages, if required.

The repository siting volume is the three-dimensional mass of rock that will be used for the emplacement of the waste packages. The repository host horizon is located above the water table in the dry, unsaturated zone, consisting of volcanic tuff, to take advantage of the features of the natural barrier.

As dictated by the siting criteria, the repository emplacement drifts and perimeter main drifts will be located entirely within this volume (Figure 4-20). The repository siting volume was defined by applying the siting criteria to a computer model of the site geology (CRWMS M&O 1997e, Section 4.2 and 4.3). The defining limits include the following:

- 200-m (656-ft) overburden thickness limit
- Top of repository host horizon (minus 5 m, or 16.4 ft)
- Bottom of repository host horizon (plus 10 m, or 32.8 ft)
- 100-m (328-ft) standoff above the groundwater table
- 60-m (197-ft) standoff from Type I faults (except on the west side of Ghost Dance fault where the standoff is 120 m, or 394 ft)

Applying these limiting criteria to the geologic model defines the repository siting volume. By sectioning through this volume at various elevations and orientations, alternative repository siting options can be examined. A section through the repository design is illustrated in Figure 4-21.

The waste emplacement block will be at least 200 m (656 ft) below the surface and at least 100 m (328 ft) above the groundwater table. The lateral limits are defined by a standoff of 60 m (197 ft), except for the Ghost Dance fault which is

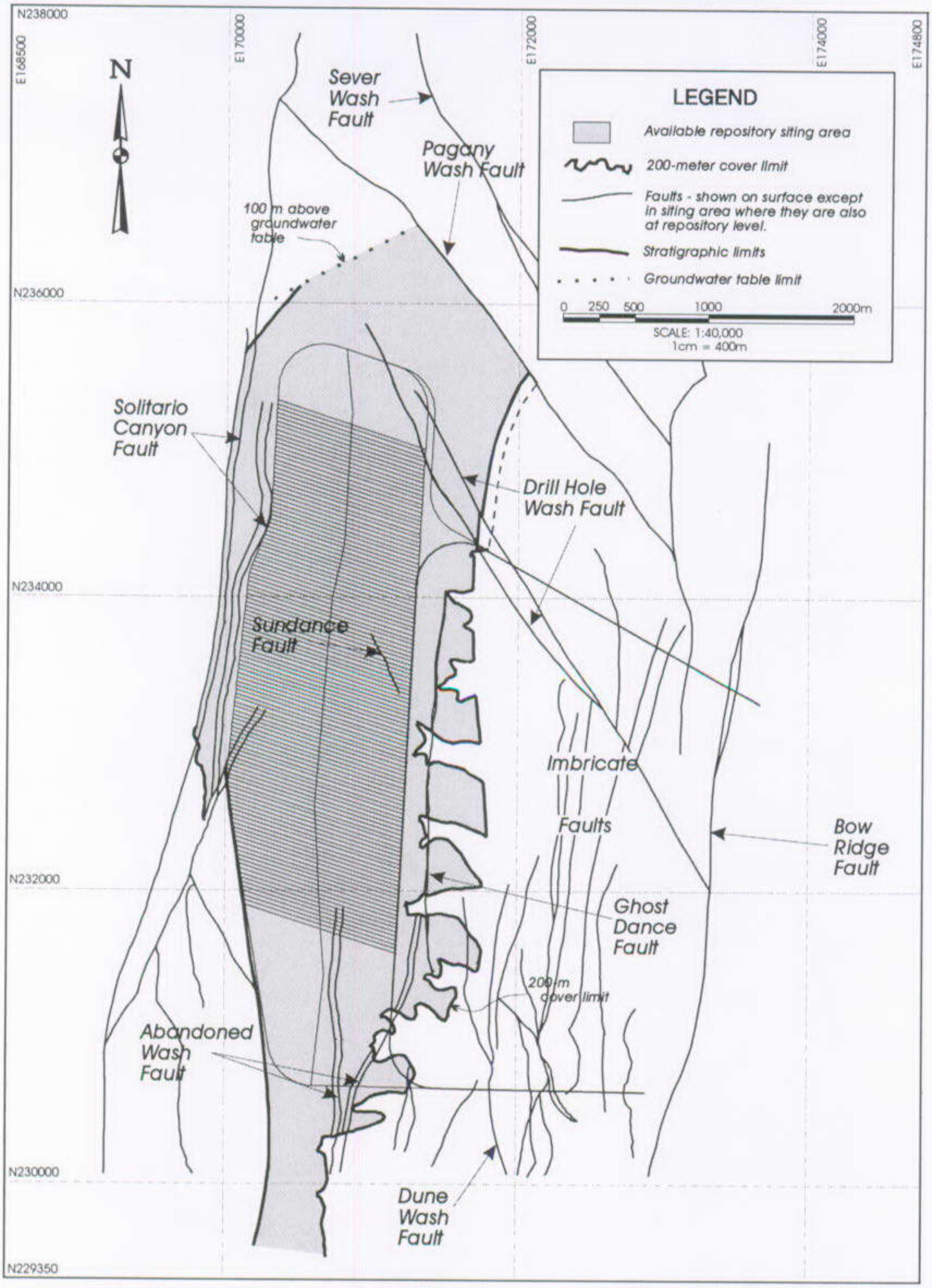


Figure 4-20. Repository Siting Limits

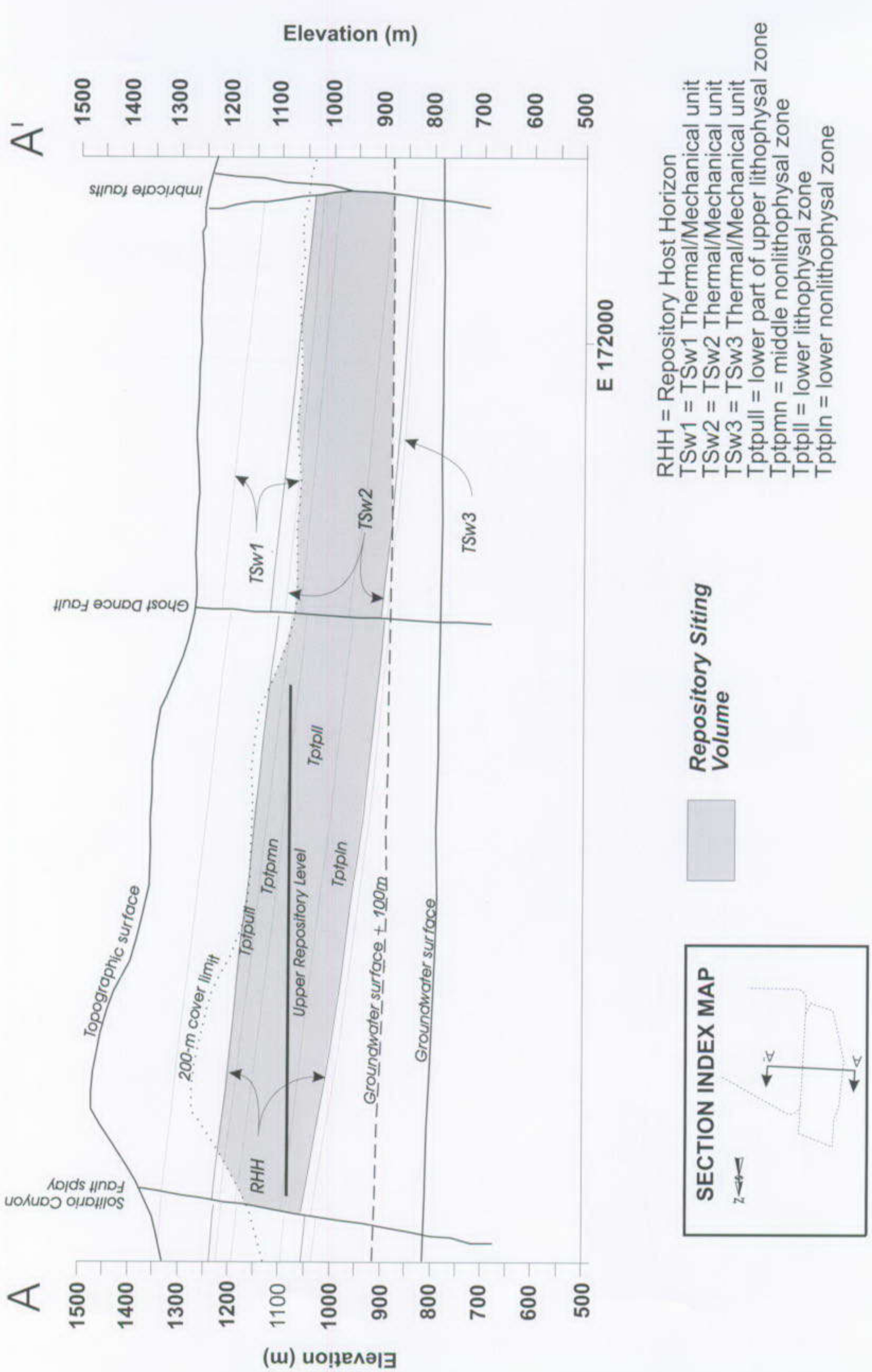


Figure 4-21. West-East Cross Section Through Repository Siting Volume

120 m (394 ft) from major faults (CRWMS M&O 1997e, Section 8).

4.2.1 Subsurface Layout

The layout of the subsurface portion of the repository is shown in Figure 4-22. Principal features identified in the figure include the north and south ramps, the east and west main drifts, the exhaust main drift located between and below the east and west drifts, and the emplacement drifts. The physical location and general arrangement of the subsurface facility in the unsaturated zone above the water table takes advantage of the mountain's natural geologic barrier and other attributes as part of the overall waste containment strategy. Another design consideration was locating the emplacement drifts away from major faults.

The subsurface portion of the repository is designed to perform the following functions:

- Accept loaded waste packages from the surface facility
- Transport the waste packages to the emplacement drifts
- Place the waste packages within the emplacement drifts
- Monitor the performance of the waste packages and drift environments before closure
- Maintain the capability to retrieve the waste packages
- Install closure barriers and seals for the underground openings, including surface and subsurface boreholes

In addition to the major design considerations affecting subsurface layout and general arrangement discussed in Section 4.2, additional considerations affecting the subsurface layout and general arrangement include the following:

- Size and weight of the waste packages: Canistered and uncanistered waste forms will be transported from the surface to the subsurface emplacement drifts in large, heavy waste packages. Section 4.2.3 describes the waste package transport system and its effect on the design of the ramps and main drifts and on the selection of waste package transportation and emplacement equipment.

- Size, configuration, and operating requirements of construction equipment: Section 6 describes the concepts for constructing and operating the repository. The construction equipment and the support functions will impact the size and shape of the excavated openings, as discussed in the *Repository Subsurface Layout Configuration Analysis* (CRWMS M&O 1997ab, Section 7.1.5).

- Construction methods, sequence, and equipment productivity: These issues affect the subsurface layout with respect to construction methods (mechanical excavation or other means) and overall constructibility. Section 4.2.1.4 discusses the construction and operation sequencing and construction methods.

- Concurrent construction and emplacement operations: Construction and emplacement operations will be performed concurrently so that the repository can begin accepting waste before all of the emplacement drifts have been completed.

- Waste retrieval after waste emplacement operations have begun: The subsurface layout must support retrieval of any one or all of the waste packages during the preclosure phase. Section 4.2.7 discusses the concepts and operations for retrieving waste packages.

The repository subsurface layout (Figure 4-22) is designed to accommodate the emplacement of 70,000 MTU as discussed in the *Repository Subsurface Layout Configuration Analysis* (CRWMS M&O 1997ab, Section 8) and the *Repository Thermal Loading Management Analysis* (CRWMS M&O 1997ac) at an areal mass

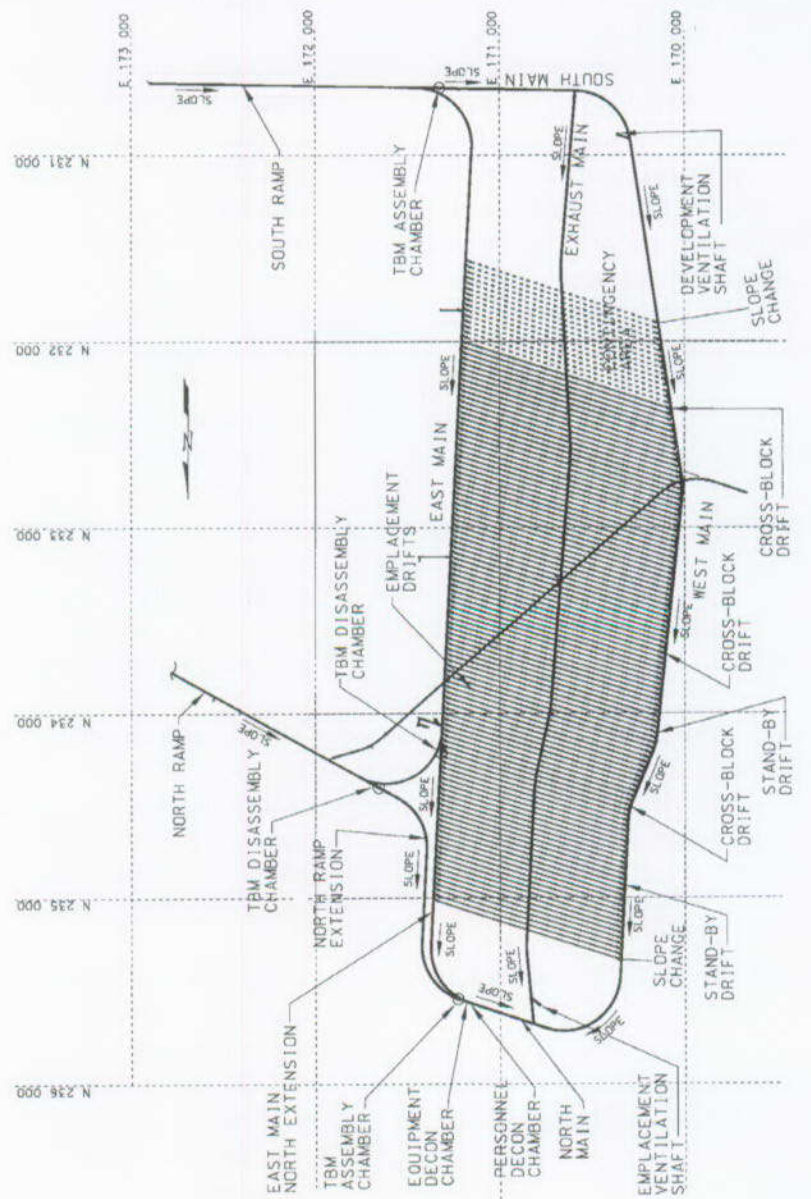


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Figure 4-22. Repository Subsurface Layout loading of 85 MTU/acre (CRWMS M&O 1997ac, Section 8). This layout will include an estimated contingency capacity of 15 percent (CRWMS M&O 1997ab, Section 7.1.3.2). The final capacity will depend on ground conditions encountered, as well as future direction by the U.S. Congress. No waste packages will be emplaced in locations where inspections have determined that the ground conditions are unsuitable. The underground area available for repository siting is described in the *Determination of Available Volume for Repository Siting* (CRWMS M&O 1997e). Flexibility in repository subsurface layout is addressed in Section 7.1.

The repository subsurface layout is configured so that most of the openings can be mechanically excavated. Access ramps, main drifts, and the emplacement drifts will be excavated using tunnel boring machines similar to the one used to bore the Exploratory Studies Facility tunnel (CRWMS M&O 1997ab, Section 4.3). A roadheader will excavate short openings such as the tunnel boring machine assembly and disassembly chambers, emplacement drift turnouts, and other drifts that cannot be practically excavated using tunnel boring machine. Although roadheader productivity is low compared to the productivity of the tunnel boring machine, the crawler-mounted roadheader has greater mobility and is better suited for excavating openings with non-circular cross sections.

The repository will incorporate the existing north ramp, east main drift (Topopah Spring main drift), and south ramp, which were excavated during development of the Exploratory Studies Facility. Exploratory Studies Facility openings will be upgraded as necessary to meet repository requirements for ground support, access, and utilities.

The major features of the subsurface repository layout are as follows:

- A series of main drifts and an exhaust main drift
- Long, parallel emplacement drifts housed in a single emplacement block
- Two ventilation shafts and two access ramps

These primary features are supported by a number of ancillary systems.

Two separate rail systems will transport workers, construction material, and the waste packages. One system will support the development area, the portion of the repository under construction. The second system, described in Section 4.2.3, will support waste transport and emplacement operations.

A final concrete invert (floor of the drift) will be installed in all openings (Section 4.2.2.5). In the ramps and mains, the final invert will be placed on top of the precast concrete invert that will be installed when the openings are constructed. There is a height differential between the emplacement drift excavation operation and the waste package emplacement operation. Because of this differential, the final invert for the emplacement drift turnouts cannot be installed until after the emplacement drift has been excavated.

Two decontamination chambers will be excavated in the north main drift: one for equipment and one for personnel. The equipment chamber will be sized to accommodate the waste package gantry mounted atop the gantry carrier, which is the largest piece of equipment required for waste package emplacement operations. The personnel chamber will be located near the equipment decontamination chamber. Both chambers will include a shower mechanism using water for decontamination. A sump will be installed in the chambers to collect contaminated water. The contaminated water will be removed from the sump and transported in containers to the Waste Treatment Building (Section 4.1.4.3) for treatment. The equipment decontamination chamber will also be equipped to provide other appropriate methods for decontamination. An air lock door will be installed at the entrance of each chamber. An independent fan system with high-efficiency particulate air filters will be installed in each chamber to pull air from the main drift into the chamber.

Two sizes of electrical alcoves will be located throughout the subsurface facility. The larger alcoves will each house a trolley rectifier, substation, and switchgear box. Each of the smaller alcoves will house a substation and switchgear box.

To prevent water from collecting around the waste packages inside the emplacement drifts, the emplacement drifts will be excavated to crest at the midpoint of the drift where the ventilation raises will be located. Any water will flow out of the emplacement drift, down the turnout, and into the east and west mains. The east and west mains will slope downward from the south to the north to the lowest point in the repository, which is along the north main where it intersects the exhaust main. Any water collected before closure will be pumped to the surface.

The utilities and systems required for excavation of the repository include nonpotable water supply, compressed air, wastewater, electrical power, ventilation, and a muck conveyor. The utilities and systems that were installed for the excavation and development of the Exploratory Studies Facility will be used wherever possible. Section 4.2.6 discusses the subsurface utility systems. Support facilities, such as Waste Handling Building and construction support facilities (Section 4.1) will be located at the surface. During the pre-emplacement construction phase, the support facilities will be located at the North Portal. During the concurrent development and emplacement operation, the facilities will be separated. The development support facility will be located at the South Portal, and the emplacement support facilities will be located at the North Portal.

Excavated chambers are required to assemble/launch and recover/disassemble, the 7.62-m (25-ft) diameter tunnel boring machine, which will excavate the access ramps, main drifts, and the exhaust main drift. The assembly and disassembly chambers for the 7.62-m (25-ft) tunnel boring machine have been located to support the desired construction sequence. Emplacement drift turnouts on the east side will serve as the launch chamber for the 5.5-m (18-ft) tunnel boring machine. Assembly chambers for the performance confirmation drifts (Section 4.2.1.2) will be excavated as needed. The 5.5-m (18-ft) tunnel boring machine, which will excavate the emplacement drifts and the performance confirmation drifts, is designed to back out of the completed drift with only minimal disassembly of some parts, so no disassembly chambers are required.

If routine repairs and maintenance cannot be provided in the drifts, the 5.5-m (18-ft) tunnel boring machine will receive required services in the underground emplacement drift turnouts. Likewise, the 7.62-m (25-ft) tunnel boring machine will receive required maintenance in the assembly or disassembly chambers if in-drift repairs and maintenance are not feasible. All other construction equipment and removable tunnel boring machine parts (from both the 7.62-m [25-ft] and 5.5-m [18-ft] tunnel boring machines) will be taken to the surface for maintenance and repairs. All waste package emplacement equipment will be taken to the surface for maintenance and repairs.

4.2.1.1 Main Drifts and Exhaust Main Drift

The east main drift was excavated using a 7.62-m (25-ft) diameter tunnel boring machine during Exploratory Studies Facility construction. The east main drift is near the top of the repository host horizon and runs parallel to the Ghost Dance fault. A similar 7.62-m (25-ft) diameter tunnel boring machine will excavate the remaining main drifts for the repository: north ramp extension, east main north extension, north main, west main drift, and south ramp extension, as discussed in the *Repository Subsurface Layout Configuration Analysis* (CRWMS M&O 1997ab, Section 7.2.1). The maximum grade in the east and west main drifts, which are used for emplacement drift access, will be 2 percent.

The exhaust main will be excavated about 10 m (32.8 ft) below the emplacement drifts using a 7.62-m (25-ft) diameter tunnel boring machine. Thus, the exhaust main cannot serve as a conduit to direct any water flow into the emplacement drifts. Each emplacement drift will be connected to the exhaust main by a 2-m (6.6-ft) diameter ventilation raise. The subsurface ventilation system is described in more detail in Section 4.2.4.

The slopes of the main drifts, exhaust main drift, and emplacement drifts, which serve to move any water seepage away from the elevated waste packages, support the waste containment strategy by limiting the amount of water that may come in contact with the waste packages, thus extending the lifetimes of the waste packages

4.2.1.2 Emplacement Drifts and Performance Confirmation Drifts (Observation Drifts)

The waste emplacement block will be at least 200 m (656 ft) below the surface and at least 100 m (328 ft) above the water table.

Locating the repository host horizon above the water table in the unsaturated zone promotes long waste package lifetimes by limiting the amount of water contacting the waste packages.

The block will cover about 300 hectares (741 acres) and will accommodate about 10,500 waste packages. The waste packages will be located along the centerline of the emplacement drifts. The emplacement drifts will run in an approximate east-west direction and be excavated using a 5.5-m (18-ft) tunnel boring machine. The emplacement drifts will be fully lined, will be accessible, and will have a maintainable service life of at least 150 years. An additional area of 107 acres has been identified as a contingency area if some drifts in the main block are determined to be unacceptable or if they are needed to accommodate higher thermal output packages (see Section 7.1).

Figure 4-22 shows 105 emplacement drifts spaced at 28 m (92 ft) center to center. Five of these drifts will remain empty during waste emplacement operations. These drifts, which are located so that they divide the block into areas of similar size, will serve two purposes.

First, three of the empty drifts will be designated as cross-block drifts and will facilitate ventilation, emergency egress, and possibly augment performance confirmation monitoring. The cross-block drifts will be the only paths for cool, fresh air to get into the exhaust main. These drifts can also function as emergency egress routes for personnel, either from the exhaust main into the repository area or from the repository area to the exhaust main, if the major routes are blocked. The cross-block drifts may also be used to monitor the adjacent emplacement drifts to support performance confirmation. However, five drifts will be developed some 15 m above the emplacement block to function as performance confirmation drifts, as discussed in the following paragraphs.

Second, the two remaining empty drifts will function as standby drifts for relocating waste packages, if needed. These two drifts will be constructed to the same standards as the emplacement drifts, but will be used only if needed. If waste packages must be removed from a loaded emplacement drift, they will be transferred to the standby drifts instead of being returned to the surface. Relocated waste packages may remain in the emplacement standby drifts permanently or until they are returned to their original locations, depending on the situation that caused the relocation.

Figure 4-22 also shows a contingency area that encompasses 15 additional emplacement drifts. This contingency area will provide an estimated 15 percent additional emplacement area. The contingency area will be used if unsuitable ground conditions are encountered within the intended emplacement areas or if additional space is needed for waste packages with exceptionally high thermal loads.

Before waste packages are placed within the emplacement drifts, the completed drifts will perform a number of functions. They will serve as ventilation airways during construction, will be used to transfer materials between the east and west mains, and will support simultaneous construction activities, such as boring ventilation and final placement of the drift inverts and utilities.

The *Subsurface Repository Performance Confirmation Facilities* analysis (CRWMS M&O 1997ag) recommends that 5 to 10 performance confirmation drifts be excavated 10–20 m (33–66 ft) above the emplacement block. The VA design incorporates five performance confirmation drifts located 15 m (49 ft) above the emplacement block. The locations of the five performance confirmation drifts are shown on Figure 4-23 as follows:

- Performance Confirmation Drift #1 is directly above Emplacement Drift #3
- Performance Confirmation Drift #2 is directly above Emplacement Drift #33
- Performance Confirmation Drift #3 is directly above Emplacement Drift #56
- Performance Confirmation Drift #4 is directly above Emplacement Drift #80
- Performance Confirmation Drift #5 is directly above Emplacement Drift #103

The performance confirmation drifts will be excavated using a 5.5-m (18-ft) tunnel boring machine and will follow the orientation and slope of the emplacement drifts. Boreholes will be drilled from the performance confirmation drifts to approach the rock mass near the emplacement drifts. Instrumentation will be installed in the boreholes to monitor conditions in the host rock. In addition, instrumentation will be placed directly in the performance confirmation drifts to monitor air temperature and relative humidity. A more thorough discussion of performance confirmation monitoring is provided in Section 4.2.5.3.

4.2.1.3 Ventilation Shafts and Access Ramps

Two shafts will be required for the subsurface ventilation system. The development ventilation shaft, located at the south end of the emplacement block, provides intake air for the development side. The emplacement ventilation shaft, located at the north end, will carry exhaust air from the emplacement operations side to the surface. As discussed in the *Repository Subsurface Layout Configuration Analysis*, both shafts will be excavated by mechanical means (CRWMS M&O 1997ab, Section 7.2.4) in multiple phases (Figure 4-24). The Repository Design Consulting Board recommended the use of mechanical excavation because the Board believes that this method would cause the least disturbance to the rock. After considering and evaluating the Board's recommendation and noting that similar methods were being used to excavate the drifts and ramps, DOE concluded in the *Subsurface Construction and Development Analysis* that mechanical excavation was indeed the better alternative (CRWMS M&O 1997af, Section 7.2.10). A raise borer will first drill a pilot hole and then back-ream a raise from the repository level to the surface. A down reamer will enlarge the raise to the full excavated diameter, and a cast-in-place concrete lining will be installed either concurrently with down reaming or from the bottom up after down reaming has been completed. The shafts will have an inside diameter of 6.1 m (20 ft), and the concrete lining will be 300 mm (11.8 in.) thick.

Excavation of the shafts will begin after the surface pads are built and access to the bottom of the shaft locations is gained. Excavation of the development shaft at the south ramp extension connector will begin only after the 7.62-m (25-ft) tunnel boring machine passes the first cross-block drift. This ensures that an alternative supply of fresh air is available for tunnel boring machine, so that the roadheader excavating the shaft connector will not contaminate the air supply to the tunnel boring machine. Similarly, excavation of the emplacement shaft/exhaust main connector will start after the 7.62-m (25-ft) tunnel boring machine finishes all excavation in the north end of the emplacement block.

The north ramp and the south ramp were excavated using a 7.62-m (25-ft) tunnel boring machine during the development of the Exploratory Studies Facility. These ramps provide access to the repository level. The north ramp is used to supply air to the emplacement side, while the south ramp provides the exhaust path for the development side. To accommodate rail transportation into the underground facility, the maximum allowable grade in the access ramps is 3 percent.

4.2.1.4 Construction and Operations Sequencing

Construction and development of the repository will be accomplished in two phases. The construction phase encompasses repository construction work that occurs before the emplacement opera-

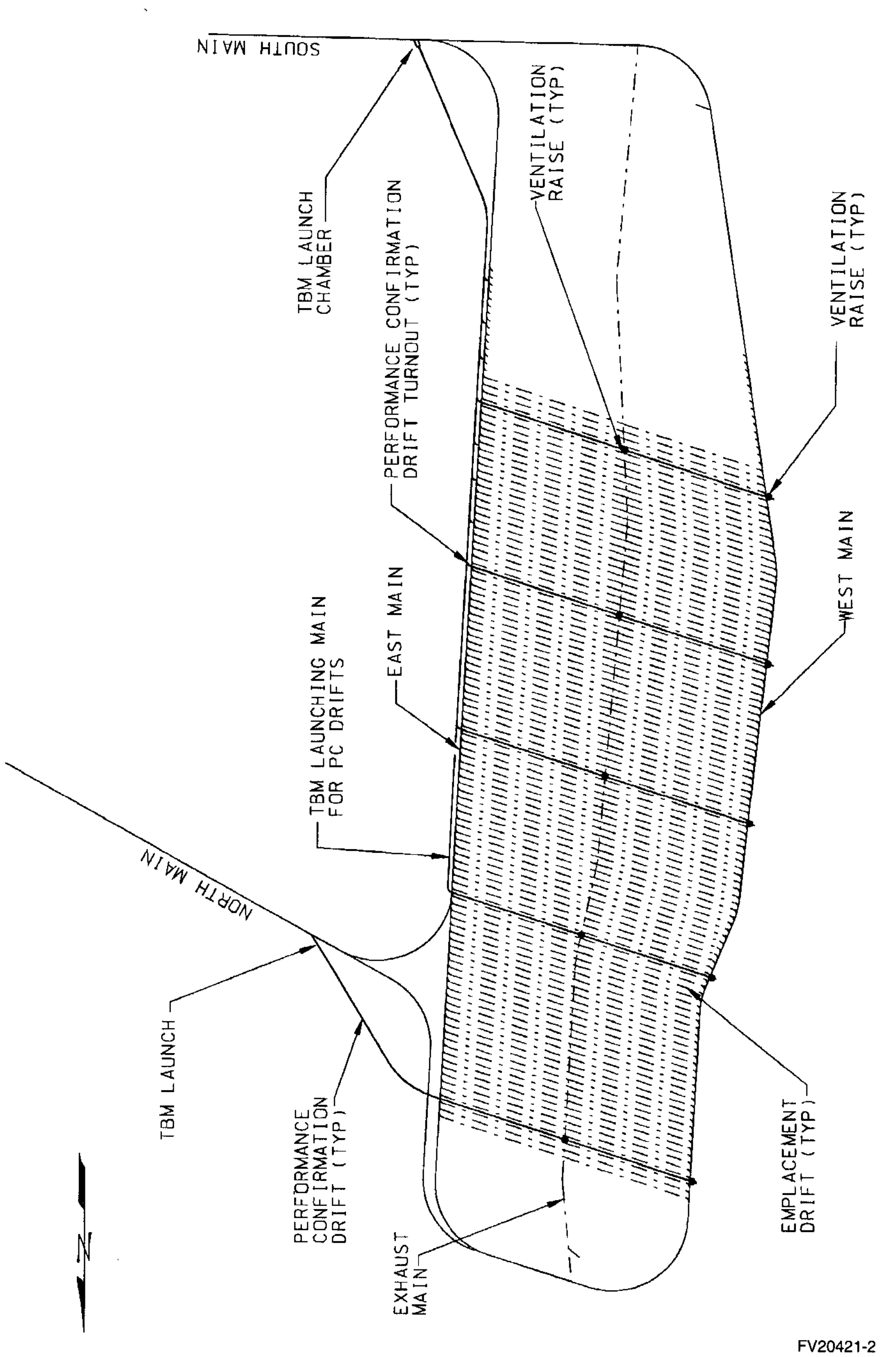

Figure 4-23. Performance Confirmation Observation Drifts

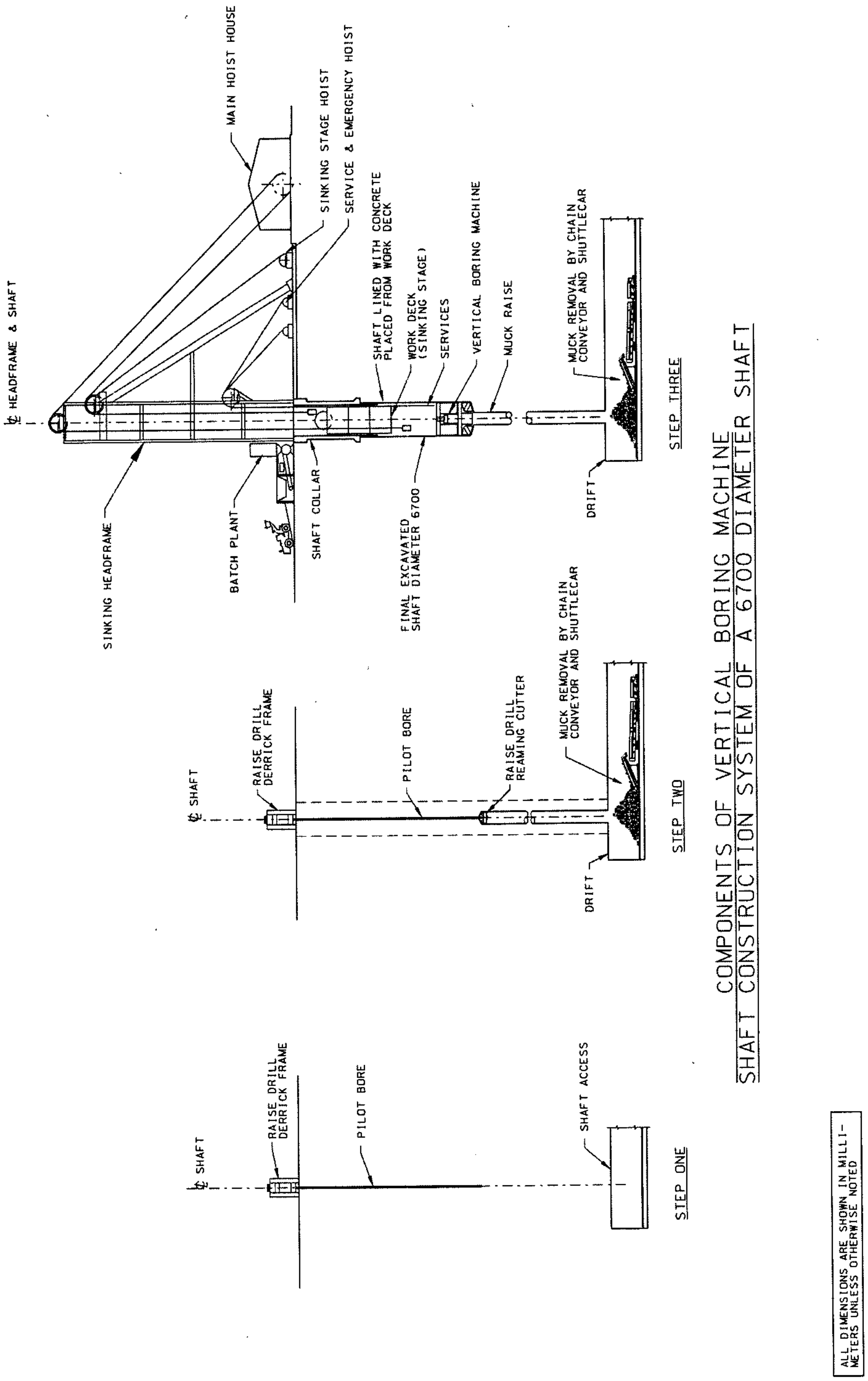

Figure 4-24. Shaft Excavation Sequence

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tions begin and includes excavation of access ramps, main drifts, ventilation shafts, and a panel of several emplacement drifts and ventilation raises. The development phase begins with the installation of the movable isolation air locks after the first panel of emplacement drifts is finished. Once the air locks are in place, the emplacement area and development area is segmented, and simultaneous emplacement and development operations can proceed.

The first feature excavated in the repository after construction is authorized will be the turnout for the southernmost cross-block drift, which is the drift just north of the contingency area shown on Figure 4-22. After the first cross-block drift is completed, the 5.5-m (18-ft) tunnel boring machine will be moved to the site of the next cross-block drift, which is near the middle of the repository. The first cross-block drift will serve as a target for aligning the west main drift along the bottom of the repository host horizon. Therefore, the first cross-block drift must be completed before the 7.62-m (25-ft) main drift tunnel boring machine reaches the curve along the south ramp extension. When the second cross-block drift has been completed, the 5.5-m (18-ft) tunnel boring machine will be moved to the last cross-block drift at the north end of the repository. The second and third cross-block drift excavations will confirm the location of the Solitario Canyon fault along the west edge of the block. The cross-block drifts will also deliver fresh air to the 5.5-m (18-ft) tunnel boring machine as it continues to excavate the mains.

The ramps and mains will be lined with concrete to ensure long life and low maintenance for the duration of the subsurface preclosure activities. Installation of the concrete lining will be scheduled to avoid interfering with other construction activities in the mains. After the lining has been installed in the main drifts, a roadheader will begin excavating the emplacement drift turnouts. After several of the turnouts have been excavated, the 5.5-m (18-ft) tunnel boring machine will begin excavating the first panel of emplacement drifts at the north end of the emplacement block. The tunnel boring machine will start from the turnouts along the east main drift and excavate across the block into the turnouts on the west main drift. The 5.5-m (18-ft) tunnel boring machine will then be partially disassembled and backed out of the lined emplacement drift to begin excavating the next emplacement drift off the east main drift.

Emplacing 70,000 MTU requires 107,144 m (66.6 miles) of emplacement drift space (104 drifts). A total of 105 drifts (108,003 m or 67 miles) will be excavated. The contingency area of 15 drifts will be excavated if the need arises. From the *Repository Subsurface Layout Configuration Analysis*, the total emplacement drift length available, including the contingency area, would be 116,957 m (72.7 miles), or 120 drifts (CRWMS M&O 1997ab, Section 7.1.3.2).

The total excavation requirements for the 5.5-m (18-ft) diameter tunnel boring machine are:

120 emplacement drifts (includes 3 cross block drifts and 2 standby drifts)	116,957 m (72.7 miles)
5 performance confirmation drifts	10,436 m (6.5 miles)

From the *Repository Subsurface Layout Configuration Analysis*, the total excavation requirements for the 7.62-m (25-ft) diameter tunnel boring machine are as follows:

North Ramp	2,804 m (1.7 miles)
East Main Drift	2,850 m (1.8 miles)
South Ramp	2,223 m (1.4 miles)
North Main Drift	1,137 m (0.7 mile)
West Main Drift	3,486 m (2.2 miles)
Exhaust Main Drift	5,048 m (3.1 miles)
East Main North Extension	1,576 m (1.0 mile)
North Ramp Extension	1,491 m (0.9 mile)
South Ramp Extension	2,269 m (1.4 miles)

(CRWMS M&O 1997ab, Section 7.2.1)

4.2.2 Ground Control System

The information in this section addresses aspects of the NRC Key Technical Issues of *Structural Deformation and Seismicity* (NRC 1997a), *Evolution of the Near-Field Environment* (NRC 1997b), *Repository Design and Thermal-Mechanical Effects*

(NRC 1997c), and *Radionuclide Transport* (Sagar 1997). A description of this key technical issue is located in Section 4.3.3.6 of Volume 4. The results of the seismic design working group have been incorporated into the VA reference design. DOE is in the process of testing the ground control system identified in this section, with the results to be incorporated into the design for the LA. Section 3.2.1.2 of Volume 4 contains a discussion of the remaining ground control work to be accomplished.

The ground control system will consist of the structures and components installed in the excavated openings to reinforce the rock surrounding the opening. This section describes the initial and final ground control systems for the following repository's major subsurface features:

- Access main drifts and exhaust main drift
- Emplacement drifts and performance confirmation drifts
- Ventilation shafts and access ramps
- Turnouts, chambers, and alcoves

These excavated features are described in Section 4.2.1.

The ground control system design process incorporated analytical results, observations, and lessons learned during construction of the Exploratory Studies Facility. Other issues considered during ground control system design included the following:

- Personnel safety
- Constructibility
- Maintenance
- Geologic mapping
- Performance confirmation testing
- Waste isolation

In addition, stress-controlled modes of failure were examined for the following:

- Excavation effects (i.e., stress change caused by excavation)
- Thermal effects (i.e., caused by heat output from waste packages)
- Seismic effects (i.e., resulting from potential earthquake events)

The ground control system designs were then incorporated into computer models for evaluating performance under in situ, thermal, and seismic loading conditions.

The initial ground control system will be installed to provide worker safety until the final systems are installed. The following three types of support are appropriate for initial ground control:

- Rockbolts (supplemented with welded wire fabric and channels)
- Steel sets (supplemented with welded wire fabric and steel lagging)
- Shotcrete (sprayed concrete that may be reinforced with welded wire fabric or steel fibers)

With the exception of the emplacement drifts, the initial ground control system will consist of rockbolts installed in a regular pattern with welded wire fabric. Shotcrete may be applied at intersections, depending on the ground conditions and the size of the roof span. Shotcrete may also be required in the main drifts to support areas of localized raveling that have been secured with rockbolts. Installation of steel sets will be required in some areas, especially for tunnel boring machine excavations, to address difficult geologic conditions (e.g., soft or fragmented ground), as were sometimes encountered during the excavation of the Exploratory Studies Facility.

4.2.2.1 Main Drifts and Exhaust Main

The conveyor system, temporary ventilation ducts, and utilities will be removed after each main drift is excavated. The initial ground control system will be installed as excavation advances to support initial tunnel excavation and mitigate potential rockfall. Forms will be erected inside the initial supports, and concrete will be pumped behind the forms to form the final drift lining. A rail mixer car will transport the concrete from a surface batch plant to the main drifts. Before the concrete fully sets, temporary steel support fixtures will be removed from drift locations scheduled for future excavation. Final ground control for the main drifts and the exhaust main, described in Section 4.2.1.1, will consist of precast concrete inverts and 300-mm (11.8-in.) thick cast-in-place concrete linings.

4.2.2.2 Emplacement Drifts and Performance Confirmation Drifts

The *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996b) recommended that rockbolts and reinforced shotcrete be used for the sides and crowns (roof) of the emplacement drifts, and that the drift invert be covered with a layer of crushed tuff. With the *Drift Ground Support Design Guide*, this concept has since changed to meet performance confirmation requirements that call for emplacement drifts to be accessible and maintainable for a service life of at least 150 years (CRWMS M&O 1997h, Section 4.2.2). This requirement was imposed so that the emplacement drifts could be periodically remotely inspected to monitor the performance of the waste packages and associated structures, including ground control systems.

The high temperatures and high radiation levels that will exist within the emplacement drifts pose many uncertainties regarding monitoring and maintenance. To address these uncertainties, DOE, in consultation with the Repository Design Consulting Board, decided to design a full lining system, which includes concrete or steel inverts. This system would be suitable for all ground conditions and construction procedures and would meet the expected requirements for waste placement and retrieval, including low maintenance.

The following three types of final ground control systems were selected for the VA design: precast concrete segmental lining, cast-in-place concrete lining, and steel sets with steel lagging. These concepts, which are described in the following paragraphs, were developed on the basis that a full lining-type ground control system would be necessary to achieve the 150-year service life with minimal or no routine maintenance:

- **Precast Concrete Segmental Lining.** A precast concrete segmental lining, illustrated in Figure 4-25, will consist of individual portions of the final lining ring prefabricated above ground, transported into the drift, and erected during drift excavation. The segments will be manufactured so that they can be assembled into an approximate full ring inside the tunnel boring machine tail shield and then moved outside the shield and expanded to form the full ring as the shield advances. Once a series of rings are in place against the rock, the gaps between the lining and the rock and within the rock itself (fissures, joints, etc.) will be filled with grout, completing that section of the lining. A precast lining can be installed rapidly and can be fabricated under controlled conditions to enhance quality and composition.

- **Cast-In-Place Concrete Lining.** This type of lining, illustrated in Figure 4-26, will be constructed by placing concrete into forms that will be either built in place or moved into place after the tunnel boring machine has been removed, unless the drift diameter is large enough to allow the removal of excavated material to proceed simultaneously with concrete work. Initial support of the drift crown (and sometimes the sides) will probably be required from the time that a section of the drift is excavated until the concrete lining is placed and set. The initial support will typically be encased in the concrete. Grouting will be performed where necessary to fill voids at the concrete-rock interface. This lining option requires that initial support, such as rockbolts and wire mesh, be installed. Although a cast-in-place concrete lining will be a viable option for the cross-block drifts and performance confirmation drifts, this option has been determined by the *Repository Ground Support Analysis for Viability Assessment* to be unsuitable for the emplacement drifts (CRWMS M&O 1997y, Section 8.0).

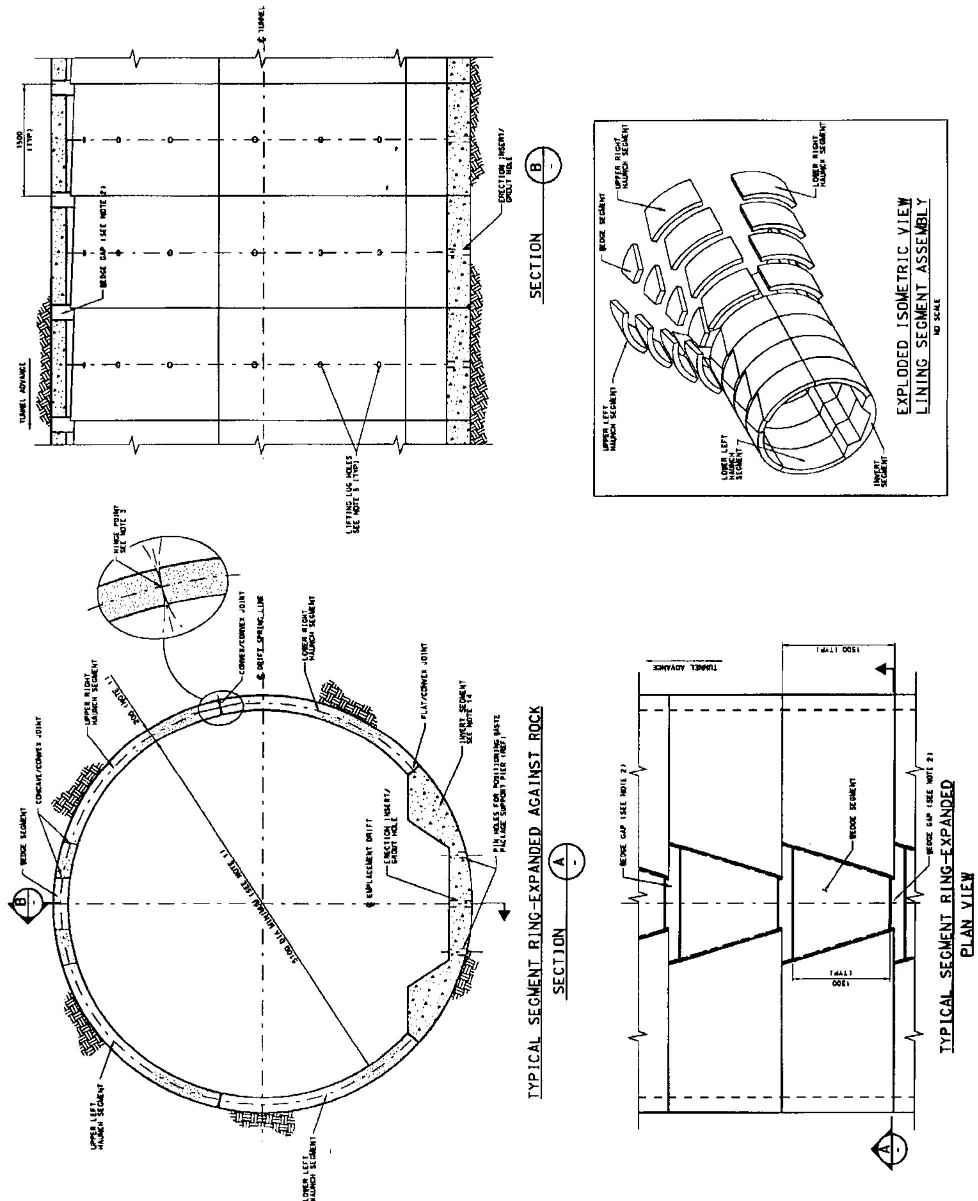

Figure 4-25. Precast Concrete Segmented Lining

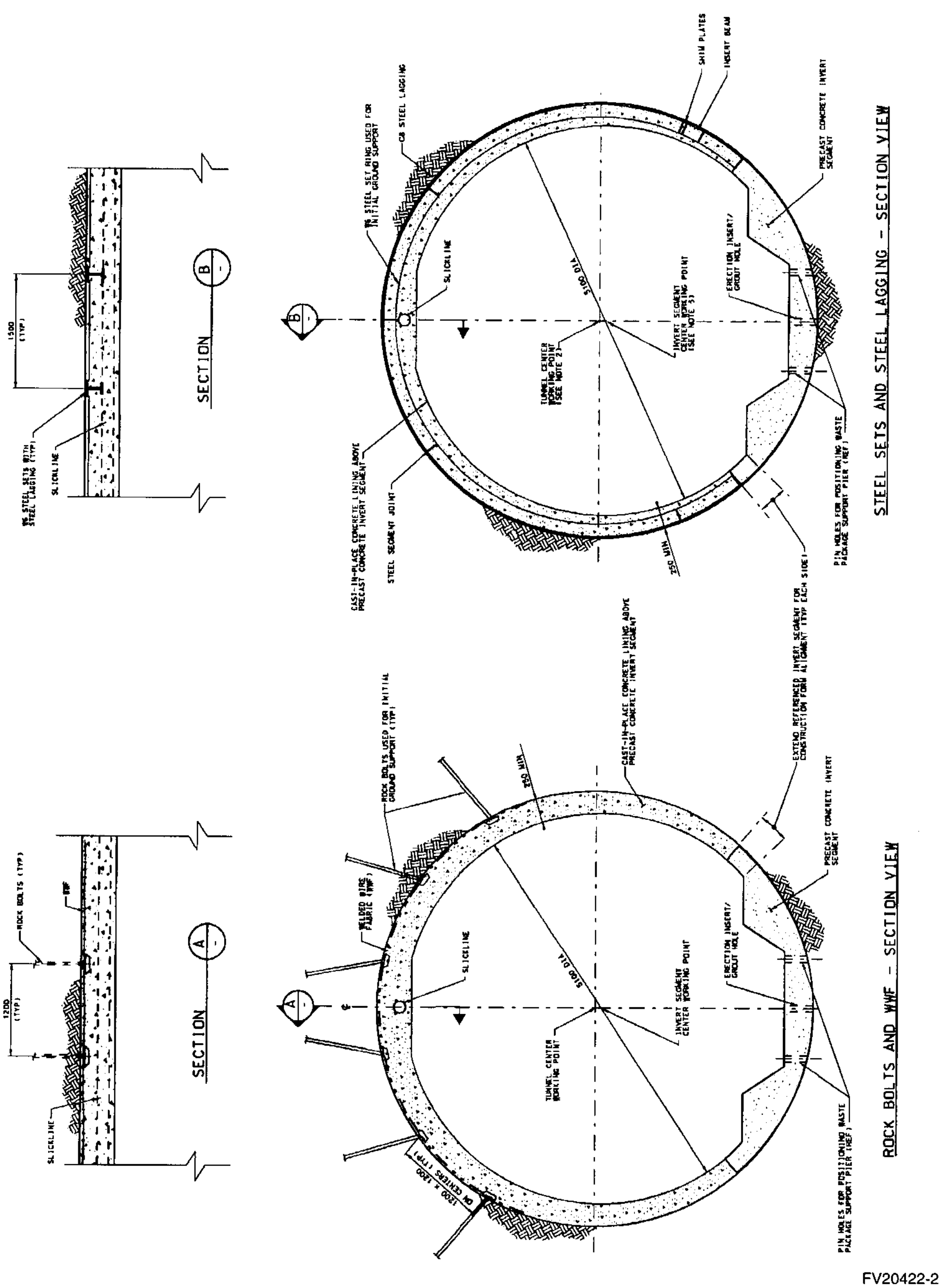

Figure 4-26. Cast-in-Place Concrete Lining

- **Steel Sets Lining System.** Steel sets, illustrated in Figure 4-27, are curved steel beams that approximate the curvature of the excavated drift. They will be assembled under the tunnel shield, moved into place as the tunnel boring machine advances, and expanded against the rock. The steel sets can be full rings installed simultaneously with either steel inverts or precast invert segments, or, an invert can be cast in place after installation of the sets. The steel sets may also be partial rings, which will be installed after cast-in-place or precast inverts will be placed. With the full- and partial-ring options, steel sets and lagging (steel plates that fit between the sets to keep rocks from falling into the excavated drift) can be installed in a single operation, or the steel sets can be installed as an initial step and the lagging installed in a later step to allow geologic mapping.

The following two major issues prompted the VA subsurface design to carry multiple options:

- The strategy for geologic mapping of the emplacement drifts

- The potential impact of the widespread use of cement materials in the emplacement environment on long-term performance

As discussed in the *Repository Ground Support Analysis for Viability Assessment*, the VA reference design incorporates precast concrete lining in 90 percent of the emplacement drifts, and steel sets and steel lagging in the remaining 10 percent. The precast concrete segmental lining will be installed in one pass in all of the emplacement drifts lined with precast concrete. In the remaining 10 percent of the emplacement drifts, the steel set lining system will be installed in two passes to allow geologic mapping of the emplacement drifts (CRWMS M&O 1998k, Section 4.3.12). This method is based on the assumption that the performance assessment analysis will conclude that the use of concrete is acceptable. However, if the analysis determines that the use of concrete is not acceptable, steel set lining systems can be installed in all of the emplacement drifts (see Volume 3 Section 3.3.2.2).

No waste packages will be placed in the cross-block drifts (Section 4.2.1.2). The cross-block drifts will be used for geologic mapping. The VA reference design calls for cast-in-place concrete to be installed as the final ground control system after the completion of the mapping activities.

The two standby drifts (Section 4.2.1.2) will also be used for geologic mapping. Since these drifts may be used for waste emplacement, they will be lined to the same standards as the emplacement drifts. A steel lining system will be installed as the final ground control system in these two drifts.

The performance confirmation drifts (Section 4.2.1.2) will not be used for waste emplacement; therefore, they will not receive the same final ground control system as the emplacement drifts. An initial ground control system consisting of rockbolts with welded wire fabric will be installed in the performance confirmation drifts. A final ground control system has not yet been selected.

4.2.2.3 Ventilation Shafts and Access Ramps

The emplacement-side ventilation shaft will be located at the north end of the waste emplacement block, and the development-side ventilation shaft will be located at the south end of the emplacement block (Section 4.2.1.3). Both shafts will be 6.7 m (22 ft) in diameter. A 300-mm (11.8-in.) thick cast-in-place concrete lining will be installed in each shaft for final ground control.

Each of the ventilation raises between the emplacement drifts and the exhaust main will be 2 m (6.6 ft) in diameter and lined with 150-mm (5.9-in.) thick cast-in-place concrete to provide final ground control.

The access ramps completed during construction of the Exploratory Studies Facility tunnel provide access to the portion of the tunnel that was excavated in the repository host horizon. As discussed

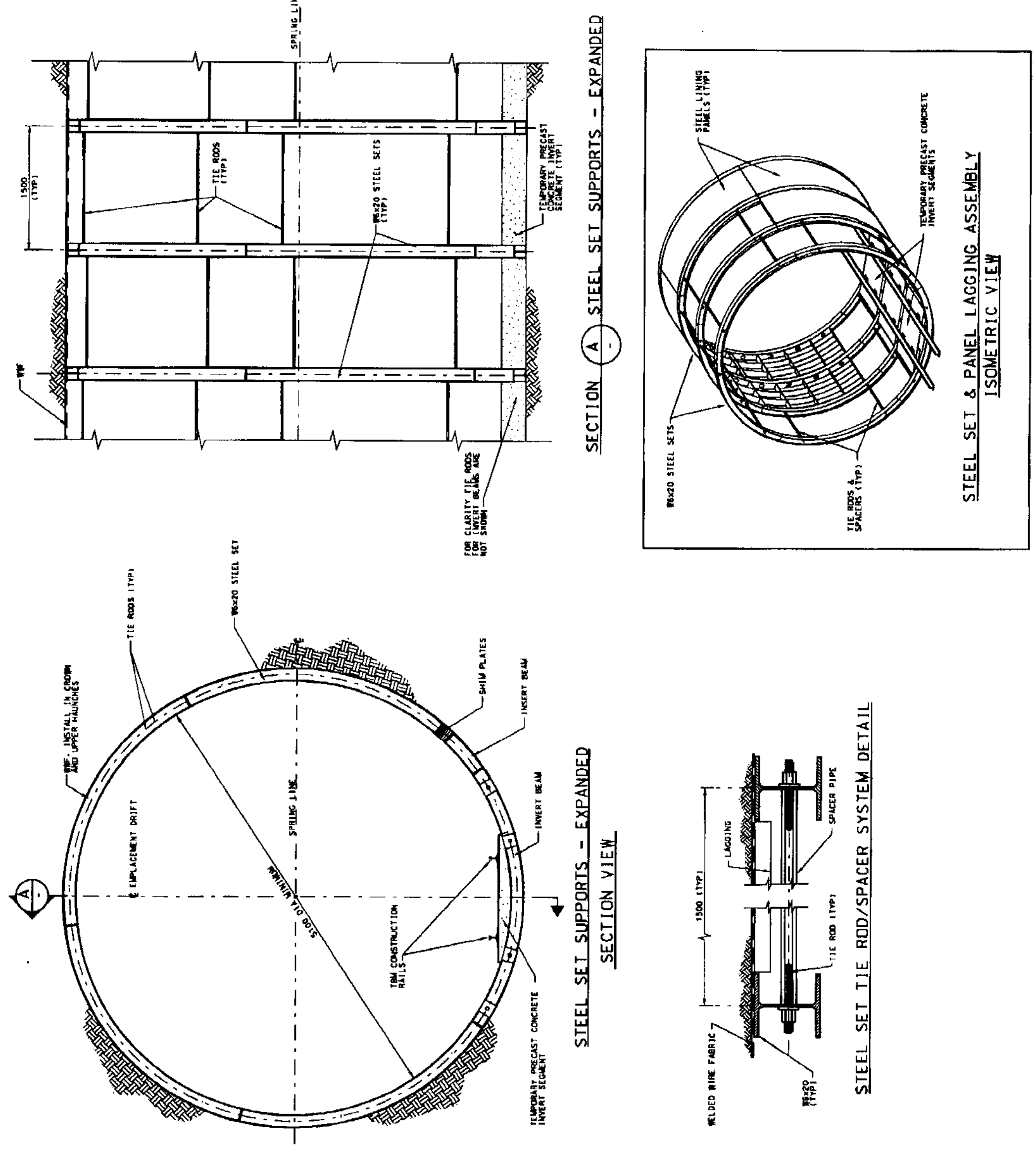

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Figure 4-27. Steel Sets Lining System

in the *Repository Ground Support Analysis for Viability Assessment*, the access ramps are supported mainly by rockbolts (3 m [9.8 ft] long) and welded wire fabric (75 by 75 mm [3 by 3 in.] wire mesh) (CRWMS M&O 1998k, Section 7.7.1.1). A 300-mm (11.8-in.) cast-in-place concrete lining will be installed as the final ground control system in the access ramps.

4.2.2.4 Emplacement Drift Turnouts, Launch/Recovery Chambers, Decontamination Chambers, and Alcoves

The emplacement drift turnouts, tunnel boring machine launch/recovery chambers, decontamination chambers, and the various alcoves required to support repository construction are described in Section 4.2.1. The turnouts will be constructed at each end of the emplacement drifts to provide space for launching or recovering the 5.5-m (18-ft) tunnel boring machine. All turnouts will be lined with cast-in-place concrete. The launch and recovery chambers will be excavated by a roadheader to provide areas for launching and recovering the 7.62-m (25-ft) diameter tunnel boring machine. Cast-in-place concrete linings will be installed in the chambers to provide final ground control. Cast-in-place concrete linings will also be installed in the equipment and personnel decontamination chambers and alcoves for final ground control.

4.2.2.5 Emplacement Drift Inverts

As described in Section 4.2.1, a final concrete invert will be installed in each emplacement drift. Where the invert structure forms part of the ground control system, it will support the rock load induced by drift excavation and the expansion of the wall and crown support elements. As part of the ground control system, the invert structure will also carry the thermal loads induced by gradual warming from decaying waste inside the waste packages and by rapid cooling (blast cooling) in the event of waste package retrieval or emplacement drift repair.

As with other components of the ground control system, stress-controlled modes of failure were examined for excavation and thermal effects. The invert design will be incorporated into the computer models to assess their performance under in situ, thermal, and seismic loading. The computer model will induce seismic loads from the design earthquake on the invert, both before and after waste placement and before and after heating and cooling.

Two types of emplacement drift invert structures were considered for VA design. Concrete and steel were both considered because of potential performance assessment concerns stemming from extensive use of concrete, which may, over long periods of time, change the pH of water in the emplacement drifts. These materials can be maintained for a service life of 150 years and can be repaired or possibly replaced if necessary. Figure 4-28, shows the concrete invert, and Figure 4-29, shows the steel invert.

During excavation of the emplacement drift, the invert structure will be installed behind the tunnel boring machine head and will support the tunnel boring machine construction rail. The invert must accommodate loads from the tunnel boring machine trailing gear, and rail traffic, for muck removal and material handling. After the emplacement drift has been excavated, the tunnel boring machine will be partially disassembled and backed out of the drift.

The emplacement drift concrete invert will form the support structure for the waste package support system. If steel inverts are installed, the waste package support assembly will be placed directly on the drift invert. Numerous materials are being tested for structural strength and long-term durability under the expected conditions of the repository for both the invert sections and the ground support.

In addition to being part of the engineered barrier system, the emplacement drift invert also provides support for the following:

- Ground support structures
- Construction access rail
- Tunnel boring machine removal by rail

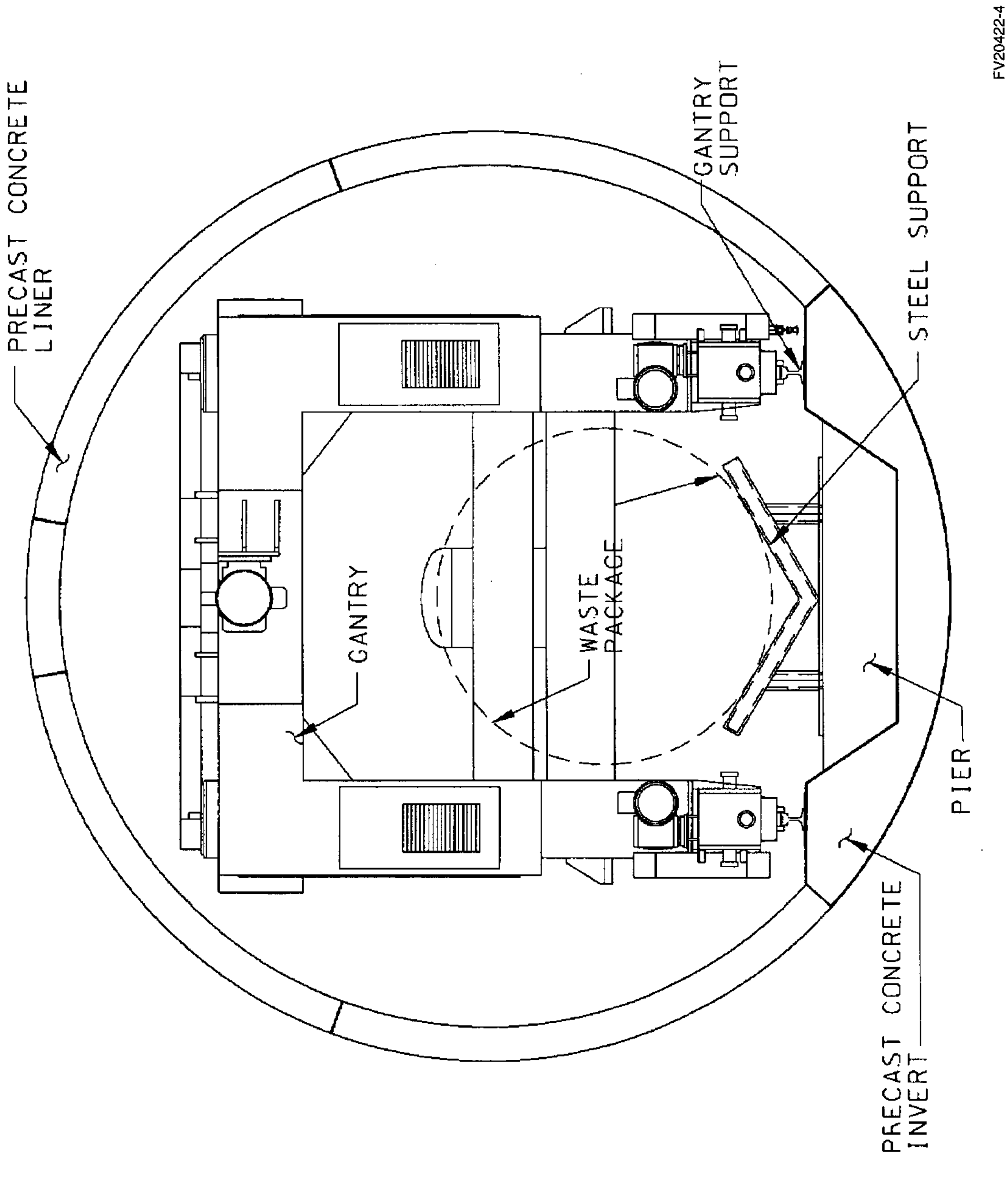

Figure 4-28. Concrete Invert

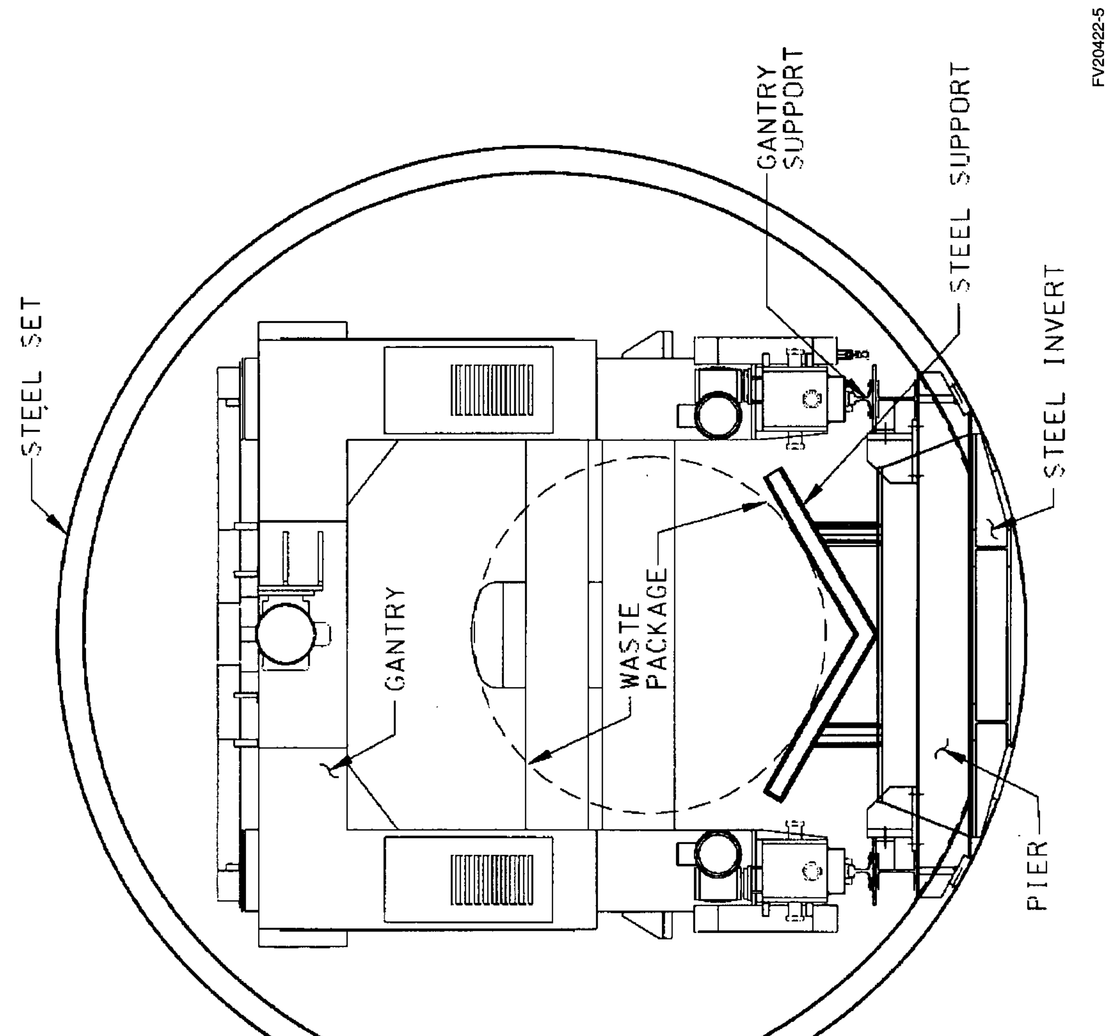

Figure 4-29. Steel Invert

- Waste package handling during emplacement and retrieval

- Waste package handling during retrieval under off-normal conditions

4.2.3 Waste Transport and Emplacement System

This section describes the mobile and stationary equipment that will be used to transport the waste packages to the subsurface and place them in the emplacement drifts. The objective of the waste transport and emplacement system design is the safe transport of waste packages to and within the repository subsurface and placement of these containers in the emplacement drifts. The waste transport and emplacement system consists of the rail transport system and the emplacement system. Each of these two components is designed to perform specific functions.

The rail transport system will perform the following:

- Receive waste packages from the waste handling building

- Safely transport waste packages to the underground emplacement drifts

- Provide shielding for workers who must perform duties around the waste package transporter

- Deliver waste packages to the emplacement drift transfer dock

- Transport the emplacement gantry from the gantry storage to the underground emplacement drifts, from drift to drift, and back to storage

The emplacement system will perform the following:

- Receive waste packages from the rail transport system at the emplacement drift transfer dock

- Transport waste packages within the emplacement drifts

- Place waste packages on support assemblies on the emplacement drift invert

Each of the two system components is made up of subsystems that are designed to execute specific tasks.

The rail transport subsystems are the rail system, locomotives, waste package transporter, reusable rail car, and overhead power system. The functions of each subsystem are described in the following list:

- The rail system will provide the access route and vehicle guidance from the surface facilities to the emplacement drifts. The rail system will also provide the electrical ground for the direct-current overhead power system in the main tunnels while the gantry rail system will serve as the electrical ground for the direct-current third-rail power system in the emplacement drifts.

- Locomotives will provide the means to move the waste package transporter or gantry carrier on the rail system. Locomotives will be equipped for both manual and remote controls. While a waste package is being transported to the transfer dock, the locomotives will be manually operated. During waste package loading and unloading, the locomotive operation will be remotely controlled to minimize worker exposure to radiation.

- The waste package transporter will provide the means to transport the waste packages underground and deliver them to the emplacement system. The waste package transporter is designed with shielding to provide radiological protection for workers and will be equipped with a rigid chain system for loading, unloading, and securing the reusable rail car during transport.

- The reusable rail car will provide a movable base for supporting and containing the waste package during transportation to the emplacement drift and serve as an interface between the waste handling building and emplacement system. The reusable rail car will travel to and from the emplacement drift inside the waste package transporter. The reusable rail car, in this application, is considered reusable compared to a previous concept of leaving the rail car in the emplacement drift as permanent support for the waste package.

- The overhead power system will provide direct-current power to the locomotives through an overhead pantograph system.

The primary emplacement subsystems will be the emplacement gantry rail system, emplacement gantry, gantry carrier, emplacement drift transfer dock and isolation doors, and shadow shield. The function of each subsystem is described in the following list:

- The emplacement gantry rail system will support and guide the emplacement gantry in the emplacement drift and serve as the electrical ground for the direct-current third-rail power system. The gantry rail will be equipped with waste package transporter unloader system guides that extend the reach of the reusable rail car unloader into the emplacement drift.

- The emplacement gantry will lift, transport, and place waste packages in the emplacement drifts.

- The gantry carrier will transport the emplacement gantry to and from the emplacement drifts. The gantry carrier will be similar to a railroad flat car equipped with rails for supporting the emplacement gantry.

- The emplacement drift transfer dock will provide an elevated platform or dock with rails and unloader guides that line up with the respective rails and guide in both the waste package transporter and the gantry carrier. The isolation doors at the transfer dock will separate the emplacement drift from the turnout area to restrict entry, control ventilation, and provide limited radiation protection.

- The shadow shield will provide radiation protection to permit personnel access in the main drift and turnout areas.

4.2.3.1 Transport System

The waste package transport equipment described in this section is based on *Preliminary Waste Package Transport and Emplacement Equipment Design* (CRWMS M&O 1997u).

Rail System. The rail system that will transport the waste packages from the waste handling building to the emplacement drifts will be a standard gauge rail (1.44-m [56.69-in.] gauge) with 115 lb/yard American Railroad Engineering Association rail. The rail system will begin with a wye configuration at the waste handling building allowing the primary locomotive to push the waste package transporter into the Waste Handling Building, where the waste package will be loaded into the transporter. The wye will also allow the primary locomotive to push the waste package transporter toward the north portal, where the secondary locomotive will be coupled. The two-locomotive configuration will then travel down the nominal minus 2.2 percent rail grade and the 305-m (1,000-ft) radius curves in the north ramp.

The rail configuration in the east main drift will include a nominal 1.4 percent grade and no curvature. The rail configuration at the emplacement drifts will be a remotely operated No. 4 rail turnout to a 20-m (65.6-ft) curve that ends at the emplacement drift transfer dock. Figure 4-30 illustrates the rail system configuration.

Locomotives. The locomotives are designed to move the radiologically shielded waste package transporter (which will contain the waste package mounted on the reusable rail car) from the Waste Handling Building, through the north portal, down the north ramp, and to the emplacement drifts. The arrangement of one of the locomotives is shown in Figure 4-31.

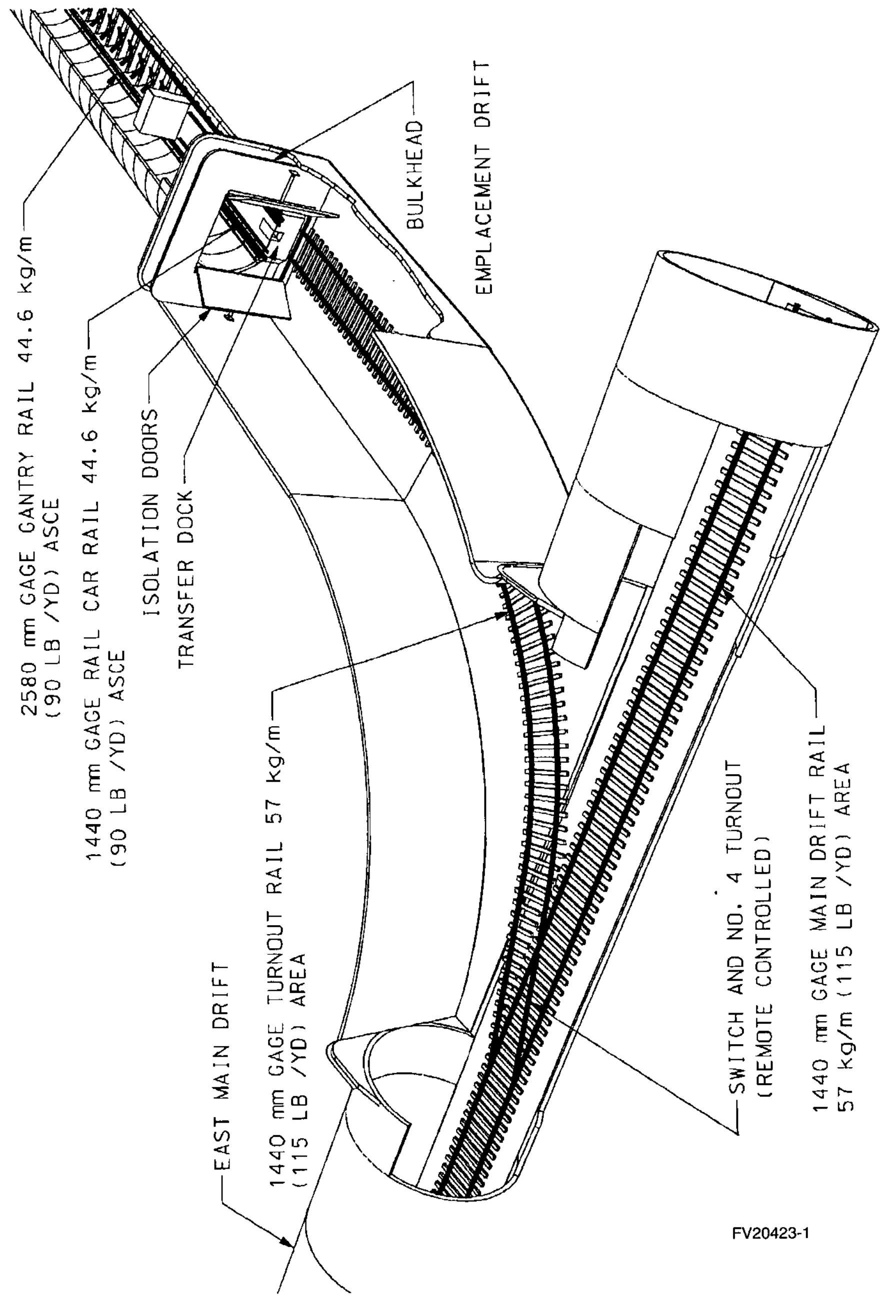

Figure 4-30. Emplacement Rail System Typical Turnout Detail

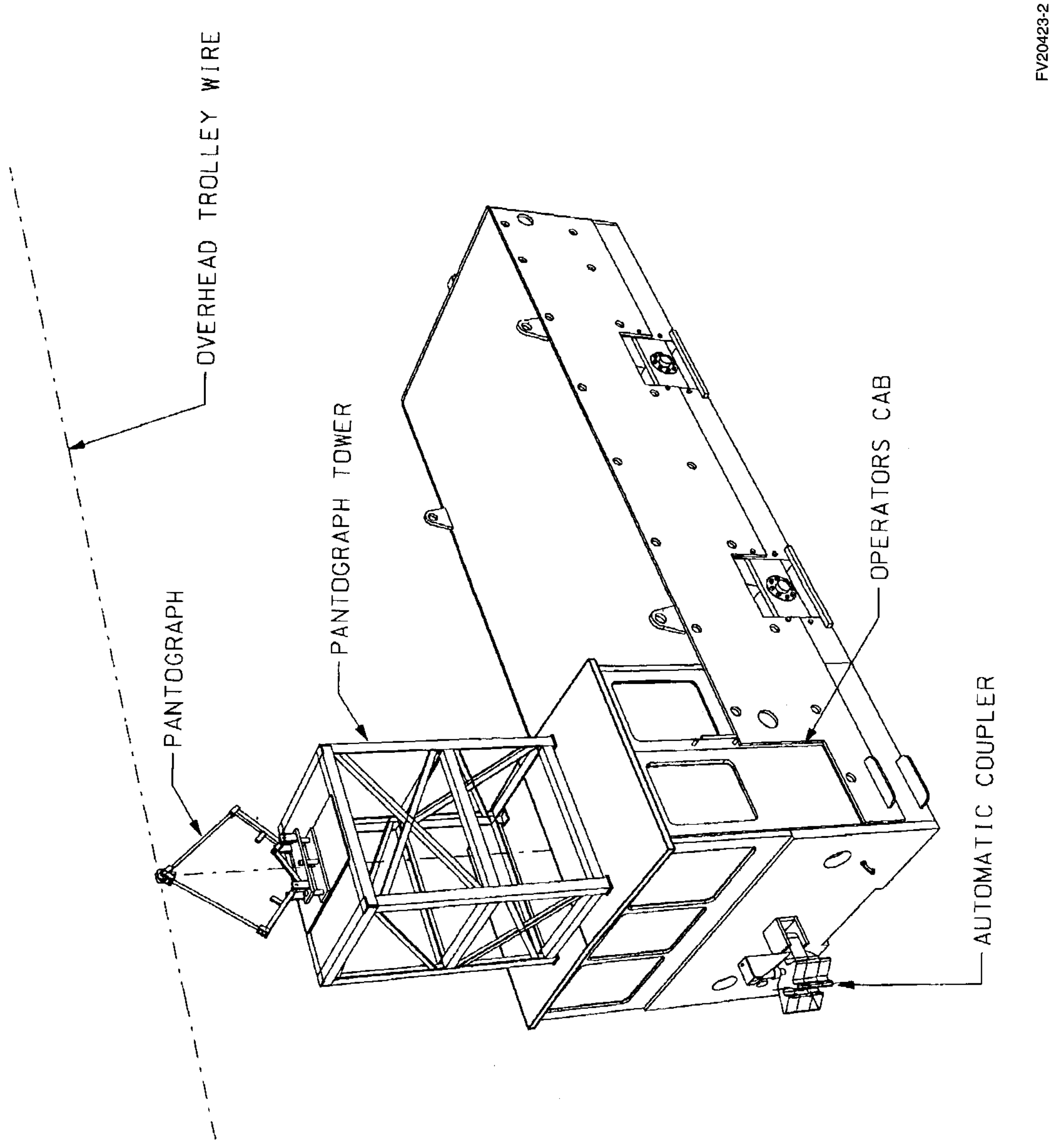

Figure 4-31. Transport Locomotive Arrangement

The waste package transporter will be moved to the emplacement drift using the following locomotive configurations:

- The primary locomotive will move the loaded waste package transporter out of the waste handling building to the north portal, where the secondary locomotive will be coupled to the unit.

- Two locomotives (both the primary and the secondary) will move the loaded waste package transporter down the north ramp.

- The primary locomotive will move the loaded waste package transporter from the main drift to the emplacement drift transfer dock, after the secondary locomotive has been uncoupled.

When placement operations in an emplacement drift cease, the gantry carrier will be coupled to the primary locomotive, which will transport the emplacement gantry to the next emplacement drift.

An overhead pantograph will be mounted on each locomotive to receive 600-volt direct-current, single-conductor power from an overhead electrical power system. The tractive power system will consist of 600-volt direct-current motors, transmissions, and wheel sets with axles, bearings, and load springs. This system will provide the horsepower necessary to move the load and will provide the reserve power needed for adequate acceleration.

The locomotives typically will be manually controlled. There may be conditions that prevent the use of on-board operators, in which case remote controls will be used. The remotely controlled functions of the primary locomotive connected to the waste package transporter will include control of the waste package transporter and its power, loading and unloading of the waste packages from the waste package-transporter, and operation of the isolation doors.

Of all transport operations, placing the waste package transporter at the emplacement drift transfer dock will require the largest-capacity locomotive, which provides the basis for locomotive design. A 45-ton-capacity locomotive has been designed for single-locomotive docking. The operation where the 45-ton-capacity locomotive will be most needed will be the movement of the waste package transporter around the 20-m- (65.6-ft) radius turnout. A specially designed locomotive has been selected to perform this duty. This locomotive has 30-in. diameter wheels on 100-in. centers and can negotiate the 20-m (65.6-ft) curve in the turnout. A standard locomotive will be modified with larger motors, additional weight, and an increased width to accommodate the 1.44-m (56.69-in.) rail gauge used in the repository.

Waste Package Transporter. The waste package transporter will move the waste packages (mounted on the reusable rail car) from the waste handling building to the emplacement drifts. A basic equipment outline and pertinent features of the waste package transporter are shown in Figures 4-32 and 4-33. As discussed in *Evaluation of Waste Package Transport and Emplacement Equipment*, the design incorporates the flexibility necessary to accommodate waste packages of varying sizes and weights up to and including a waste package that will be 2 m (6.5 ft) in diameter, 5.85 m (19.19 ft) long, and will weigh 69,000 kg (151,800 lb or approximately 76 tons) (CRWMS M&O 1997l, Section 7.2). During transport, the waste package transporter will perform the following functions:

- Shield workers from radiation

- Provide a safe, stable platform for transporting the waste package mounted on the reusable rail car

- Provide structural integrity for supporting the load, coupling to the locomotives, and braking systems for stopping and speed control

- Line up the reusable rail car rails accurately with the emplacement drift rails so that the reusable rail car can travel into the drift

- Provide a solid connection and restraint at the drift docking point

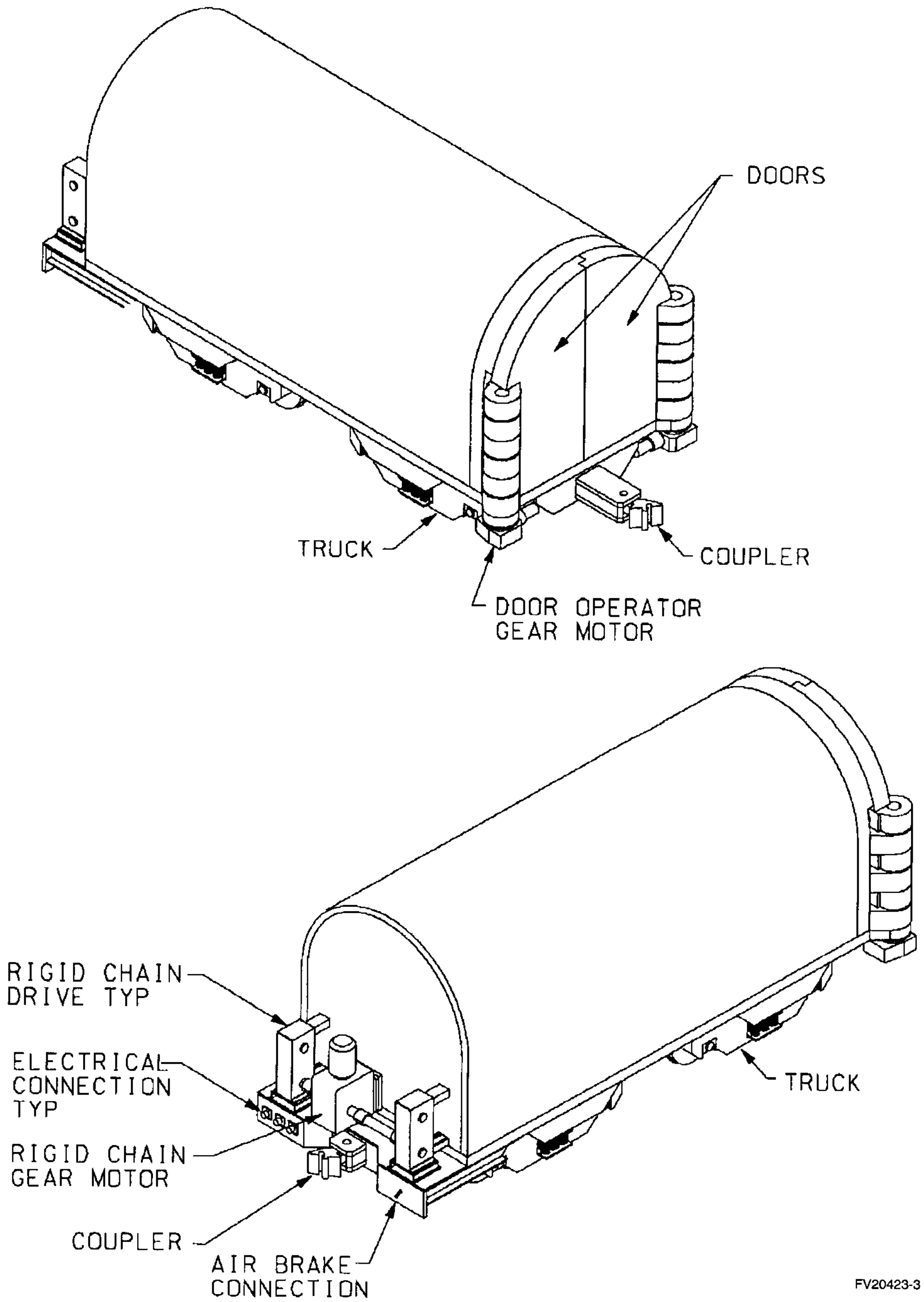

Figure 4-32. Waste Package Transporter Arrangement

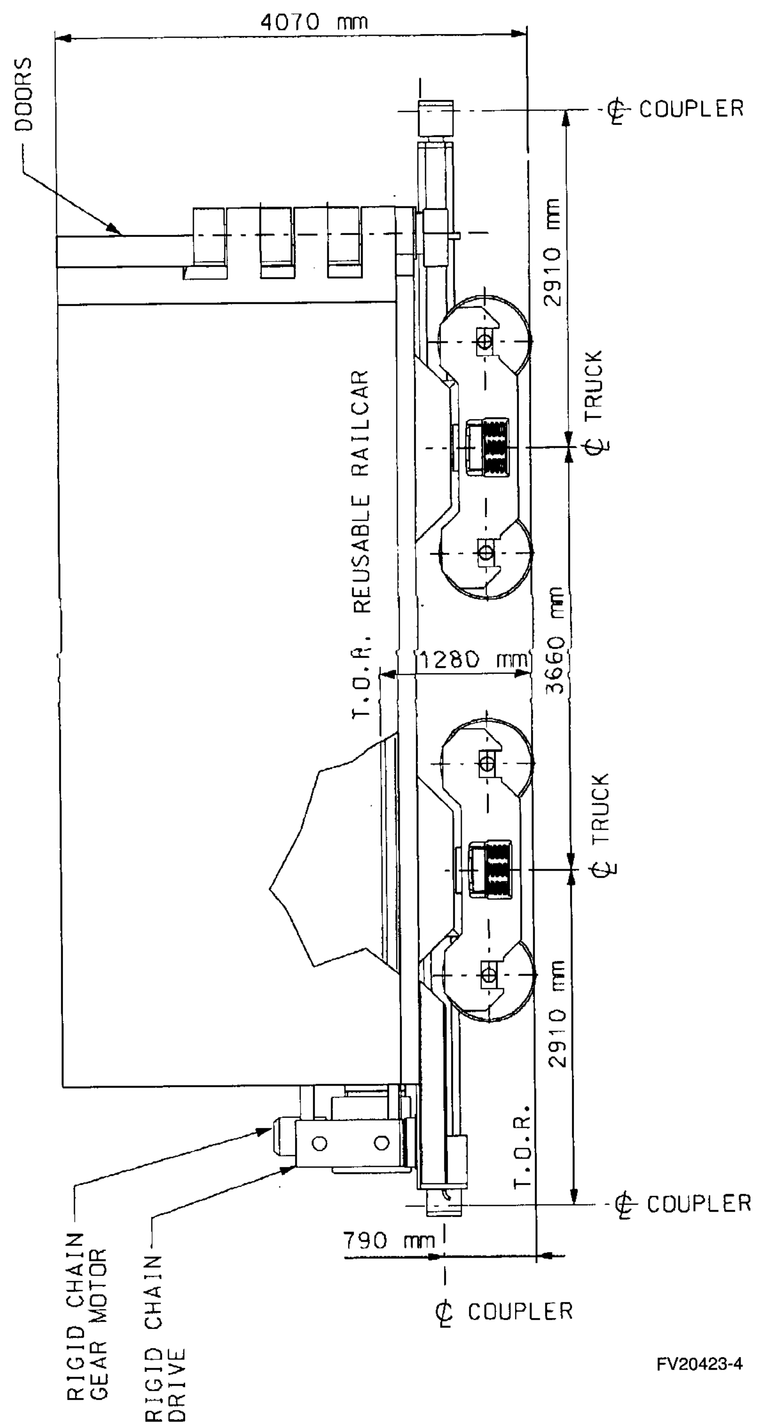

Figure 4-33. Waste Package Transporter Side Evaluation

- Provide an integrated loading and unloading system for the reusable rail car that will transfer the waste package from the transporter to the emplacement drift

- Provide mechanisms for remotely opening and closing the waste package transporter doors

- Provide manual connections to the locomotive for power, systems control, and brakes

The waste package transporter will be composed of several component systems including the shielding, underframe, undercarriage, couplers, brake system, automatic doors, reusable rail car restraint, reusable rail car unloading system, wiring, interlocks, and instrumentation. Future work as noted in Volume 4, Section 3.2.1.4 will include the waste transport system and emplacement gantry that will identify other potential waste package sizes and weights. At the time the analyses were performed, the weight of the heaviest waste package was approximately 76 tons. Subsequent to performing the analyses discussed in the VA reference design, the weight of the heaviest package has been determined to be on the order of 91 tons (for naval spent nuclear fuel). Reanalyses using the higher weight will be completed before any license application is submitted.

Shielding. Shielding is designed to reduce the radiation from the waste package inside the transporter to a level of less than 50 mrem/hr at the outside surface of the transporter, which, as discussed in *MGDS Subsurface Radiation Shielding Analysis*, is compatible with the operation requirements within the main drifts (CRWMS M&O 1997n). The shielding is a composite of stainless steel, carbon steel, and a borated polyethylene material with a total thickness of 264 mm (11 in.). The radiation shielding will be designed to protect workers from gamma and neutron radiation, and will consist of the following materials at the densities and thicknesses specified in Table 4-2.

Table 4-2. Radiation Shielding Material Densities and Thicknesses

Material	Material Density kg/m^3 (lb/ft^3)	Material Thickness mm (in.)
Borated polyethylene (1.5 percent boron)	920.0 (57.4)	101.6 (4) one layer *76.2 (3) one layer
Carbon steel	7,832.0 (488.9)	152.4 (6) one layer *177.8 (7) one layer
Stainless steel	7,949.7 (496.2)	5 (0.2) two layers

*The shielding dimensions and materials for the waste package transporter are 5 mm (0.2 in.) SS316L + 152.4 mm (6 in.) carbon steel + 101.6 mm (4 in.) borated (1.5 percent boron) polyethylene + 5 mm (0.2 in.) SS316L in the radial direction; and 5 mm (0.2 in.) SS316L + 177.8 mm (7 in.) carbon steel + 76.2 mm (3 in.) borated polyethylene + 5 mm (0.2 in.) SS316L in the axial direction. The axial shielding dimensions are derived from the radial shielding thickness by increasing the carbon steel thickness by 25.4 mm (1 in.) to account for the increased gamma radiation field from the fuel assembly end fitting sources, and by decreasing the reduced neutron field on the ends.

The carbon steel shield, which will face the inside of the waste package transporter, will provide gamma shielding and serve as the shielding support structure. The 1.5 percent borated polyethylene neutron shielding material will be attached to the outside surface of the carbon steel. The 1.5 percent borated polyethylene will be covered on the outside with stainless steel. The shield is designed as a rigid box with doors. The box will not only be self-supporting, but will also serve as a structural component of the underframe. The carbon steel inner shield material will be fabricated and machined into door hinges and other features as required.

Underframe. The underframe structure below the floor will connect the waste package transporter shield and couplers to the rail trucks. The underframe will also support the auxiliary equipment including the door operator, air brakes, reusable rail car unloader, and miscellaneous electrical equipment, controls, wiring, and instrumentation.

The underframe will be fabricated of structural steel sections and plates with both welded and bolted connections. There will be two sections of the underframes, one front and one rear, which will be connected to and integrated with each other by the underside of the shielding floor. The front underframe will include the front coupler, front bolster plate, and an equipment platform for the reusable rail car unloader. The rear underframe section will include the rear coupler, truck bolster plate, and the door operator drive.

Undercarriage. The undercarriage will be composed of rail trucks that support the underframe of the waste package transporter. The rail trucks will be a standard rail car configuration adapted to this specific application. The trucks will include wheels, axles, bearings, brakes, and springs, which will be incorporated into the truck frame, and the bolster, which will transmit the load from the truck to the bolster plate of the underframe. The truck bolster will include a bolster pin, which will center the truck in a corresponding hole in the bolster plate and allow the trucks to pivot and the waste package transporter to negotiate curved track. Truck capacity is based on the maximum operating load, which is the maximum operating weight distributed over the eight wheels of the two trucks. The maximum design operating weight determines the design of the wheel and rail combination. Wheels will have a 30-in. diameter, and 115 lb/yard rail will be used. The maximum operating weight is 233.15 metric tons (256.47 tons), and the design wheel load is 44,961 kg (98,914 lb). The wheels and rail will be similar to those used in the bridge crane service; the wheels will be heat-treated to achieve a Brinell hardness number of 615 and the rails a Brinell hardness number of 320.

Couplers. The waste package transporter will be equipped with two couplers, which are common in rail equipment used for underground mining and tunneling. This coupler configuration will allow operation of the primary locomotive at the front of the waste package transporter and the secondary locomotive at the rear during transport from the surface to the emplacement drift.

Brake System. An interconnected, fail-safe air brake system on the waste package transporter will operate in conjunction with the primary locomotive, similar to standard rail industry practice. The system will use spring-set air-release brakes and includes the waste package transporter brake shoes and air cylinders, as well as the operating linkage installed on the rail trucks. The air reservoir, piping, and other miscellaneous equipment will be located on the underframe. The air brakes will be connected to the primary locomotive with rail-industry-standard manual connections. The waste package transporter brake system will provide a redundant stopping system that will be separate from the locomotive braking system.

Automatic Doors. The waste package transporter's automatic doors will be controlled from the primary locomotive by a removable connector between the locomotive and waste package transporter. An operator will be assigned to each door. Working in unison, the operators will open and close the shielded doors for loading the reusable rail car at the waste handling building and unloading at the emplacement drift. The doors will rotate in lubricated sleeve bearings on the door hinges, allowing the doors to swing 270 degrees to the side of the waste package transporter. The door hinges will include a thrust bearing to support the vertical load. The doors will be rotated in either direction for opening or closing by a hinge pin that extends through the shield floor and connects to a low-speed motor gear reducer through a spline joint. The motor gear reducer will be a right-angle helical-worm unit flange-mounted to the underside of the shield floor. The doors will be equipped with a locking device to secure the closed doors against the body of the waste package transporter, maintaining the radiation seal and preventing accidental opening during transport. The locking device will be operated remotely and interlocked with the manual door operation.

Reusable Rail Car Unloading System. The reusable rail car and waste packages will be unloaded from the waste package transporter to the emplacement drift using the reusable rail car unloader system. The loading and unloading mechanism incorporates a "rigid chain" design; the chain coils like a conventional chain, but assumes the characteristics of a rigid bar when it uncoils and is subjected to a force along its uncoiled length. This feature will allow the chain to be stored in the coiled state, thereby providing the 12 m (39 ft) of travel required to push the reusable rail car into the emplacement drift.

The unloading system will use two separate rigid chains. The design allows all chain to be stored in magazines inside the waste package transporter on each side of the waste package. The drives will be located outside the shielding on an equipment platform to allow for service and repair. The two drives will be connected to a common gearmotor for synchronous operation. Each of the rigid chains will be drawn from the respective magazine through a penetration in the shielding to the drive and then driven back through another penetration in the shielding. Inside the shielding, the rigid chain will run through a guide installed on the floor. At the end of the guide, the chain will be connected to a "pusher bar" that will engage the rail car and move it through the open automatic doors and into the emplacement drift. Proper alignment of the tracks and the two rigid chain guides between the waste package transporter and the emplacement dock will be critical for proper operation of the reusable rail car.

Reusable Rail Car Restraint. The reusable rail car will be attached to the loading and unloading mechanism. This mechanism will position the reusable rail car on the rails both inside the waste package transporter and inside the emplacement drift. A restraint will secure the waste packages and reusable rail car when the waste package transporter is in the transit mode. The reusable rail car unloader engagement hook will prevent the reusable rail car from traveling along on the rails when the reusable rail car is in the unloading position (see Figure 4-34).

Wiring, Interlocks, and Instrumentation. The waste package transporter will be prewired for power and control. The wiring will be contained in rigid conduit to be located outside the waste package transporter shielding. Installing the conduit outside the shielding will simplify maintenance and repair. The wiring, required for interconnection to the primary locomotive, will be routed in separate cable bundles for power and control to separate terminal blocks in a common enclosure. The cables between the waste package transporter and locomotive will be connected with quick connect and disconnect connections. The control electronics will be installed inside an enclosure, which will be easily accessible for connecting the locomotive cables to the transporter cables. The waste package transporter will be equipped with interlocking devices for safety and proper operation. Instrumentation installed on the waste package transporter will monitor internal environmental conditions and relay the data back to the primary locomotive. The following waste package transporter electrical systems will be connected to the primary locomotive:

- Door system power, control, and status
- Unloader system power, control, and status
- Reusable rail car unloader connection power, control, and status
- Transfer and loading dock alignment status
- Closed-circuit television monitor power and controls
- Environmental data (e.g., temperature) connections
- Lighting

Reusable Rail Car. The reusable rail car will support and move the waste package into the waste package transporter, support and secure the waste package inside the waste package transporter, and transfer the waste package from the waste package transporter to the emplacement drift.

The reusable rail car, as depicted in Figures 4-35 and 4-34, has been designed to accommodate various sizes of waste packages. The physical sizes and weights of the waste packages to be carried by the reusable rail car are described in Section 5.1 of this volume. The design of the reusable rail car is

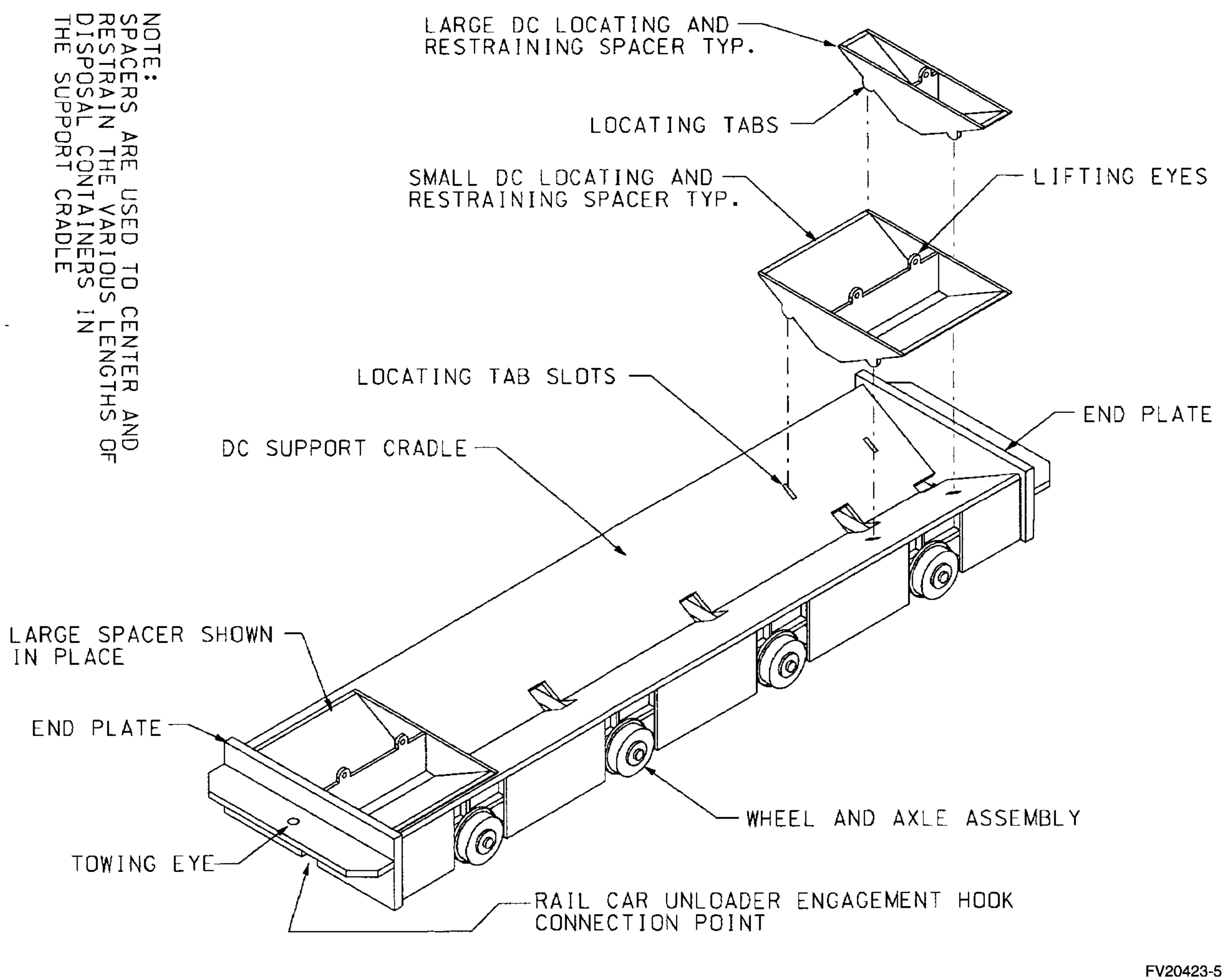

Figure 4-34. Waste Package Transporter Rail Car Unloader System

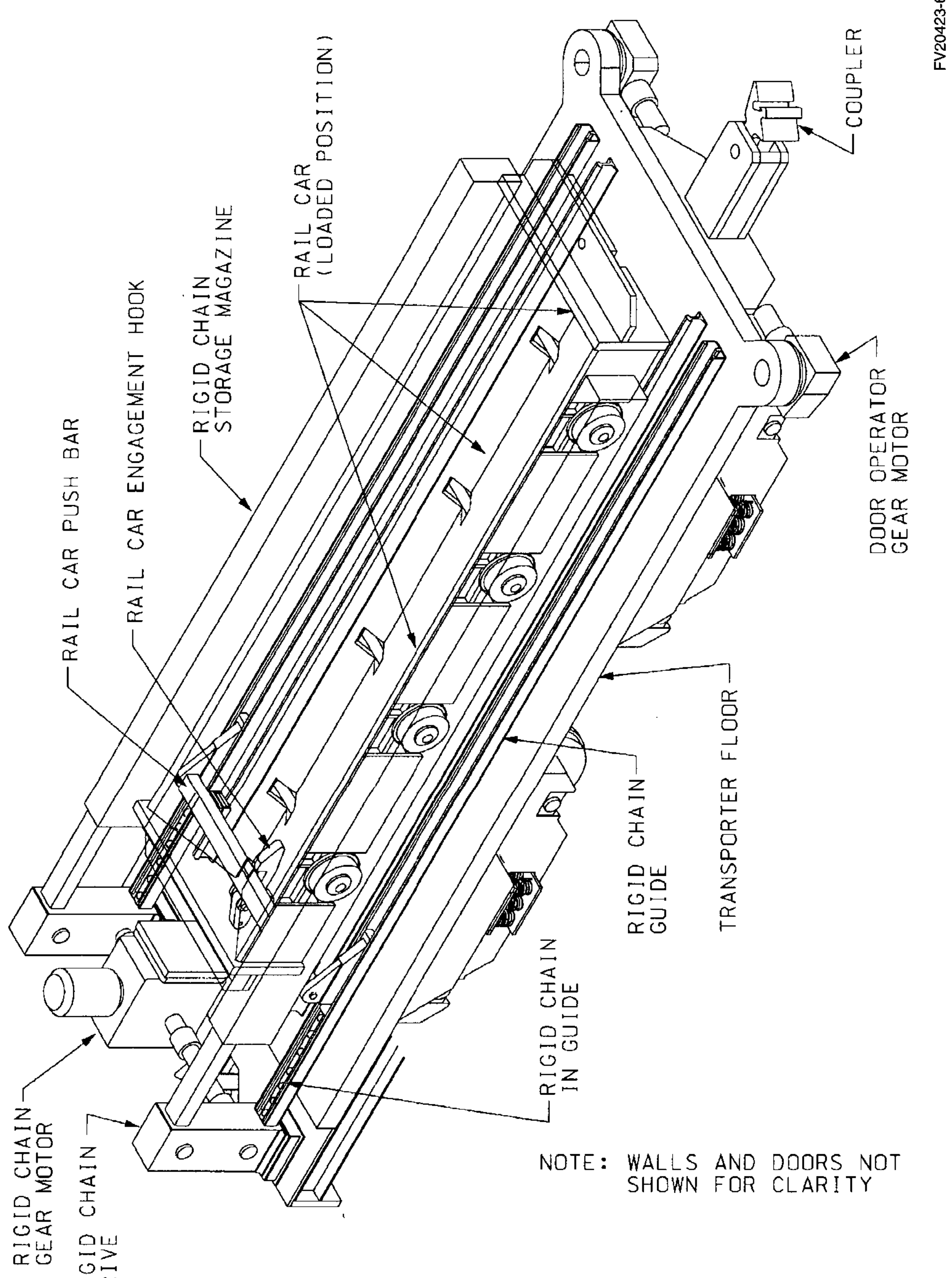

Figure 4-35. Reusable Rail Car Arrangement

based on a waste package with maximum dimensions (2 m [6.5 ft] in diameter by 5.85 m [19.19 ft] long) and a maximum weight of 69,000 kg (151,800 lb or 75.9 tons). The reusable rail car normally will be attached to the waste package transporter by the loader and unloader system. However, it may be detached from the waste package transporter as necessary for loading waste packages in the Waste Handling Building or for maintenance. The reusable rail car will be equipped with a towing eye on both ends for connecting other moving equipment when the rail car is detached from the loader and/or unloader.

The rail car will be fabricated from ASTM A36 structural steel plate welded in accordance with American Welding Society welding standard D1.1. The V-shaped waste package support, or cradle, will be supported on a boxed structural section with provisions for attaching four-axle assemblies. The length of the cradle will accommodate the longest waste package (5,850 mm [234 in.]). Shorter waste packages will be centered lengthwise between removable spacers placed at each end of the cradle. This procedure will not only prevent longitudinal shifting of the waste package while in transit, but also ensure that the wheels will be loaded equally and the waste package will be positioned for engagement by the emplacement gantry lifting heads in the emplacement drift. When required, the spacers will be installed at the Waste Handling Building with the waste package. The spacers will be equipped with locating tabs that fit into corresponding slots in the cradle, thereby positioning the spacers against each end plate on the reusable rail car.

The rail car and waste package will be supported on four equally spaced wheel and axle assemblies. Each assembly will include an axle with supporting brackets for attaching the axle to the underframe. Wheels with integral bearings will be attached to each end of the axle. The assembly will be a commercial grade product used in heavy-duty and severe applications, such as foundries, mills, and mines. The unit selected for this application uses 12-in. diameter, single-flange rail wheels with a rated capacity of 15 tons. The reusable rail car will travel on 90 lb/yard American Society of Civil Engineers-designation rail in the waste package transporter and in the emplacement drift transfer dock.

4.2.3.2 Waste Emplacement System

The waste emplacement equipment described in this section is based on *Preliminary Waste Package Transport and Emplacement Equipment Design* (CRWMS M&O 1997u).

Emplacement Gantry Rail System. Two separate rail systems will be located within the emplacement drift. The outer rail system will be a 2.85-m (112.2-in.) gauge rail designed for the emplacement gantry. The inner rail system will be designed for the reusable rail car as it is pushed into the emplacement drift. This 1.44-m (56.69-in.) gauge emplacement drift rail system will be compatible with the reusable rail car rail gauge inside the waste package transporter.

Both the emplacement gantry rail and the emplacement drift rail will use 90 lb/yard American Society of Civil Engineers rail. The 90 lb/yard rail will have a Brinell hardness number of 320 and will be able to accommodate both the 1.44-m (56.69-in.) gauge application for the reusable rail car and the 2.85-m (112.2-in.) gauge application for the emplacement gantry. Neither the emplacement drift rail nor the gantry rail will have curves.

Emplacement Gantry. The emplacement gantry will receive waste packages from the reusable rail car and position them in the emplacement drift. The emplacement gantry, shown in Figures 4-36, 4-37 and 4-38, will be a self-propelled, remotely operated vehicle. The emplacement gantry will perform the following functions:

- Accurately position itself over the waste package on the reusable rail car to ensure engagement of the waste package skirts by the emplacement gantry lifting heads
- Lift the waste package from the reusable rail car and over the reusable rail car end plate and the concrete shadow shield

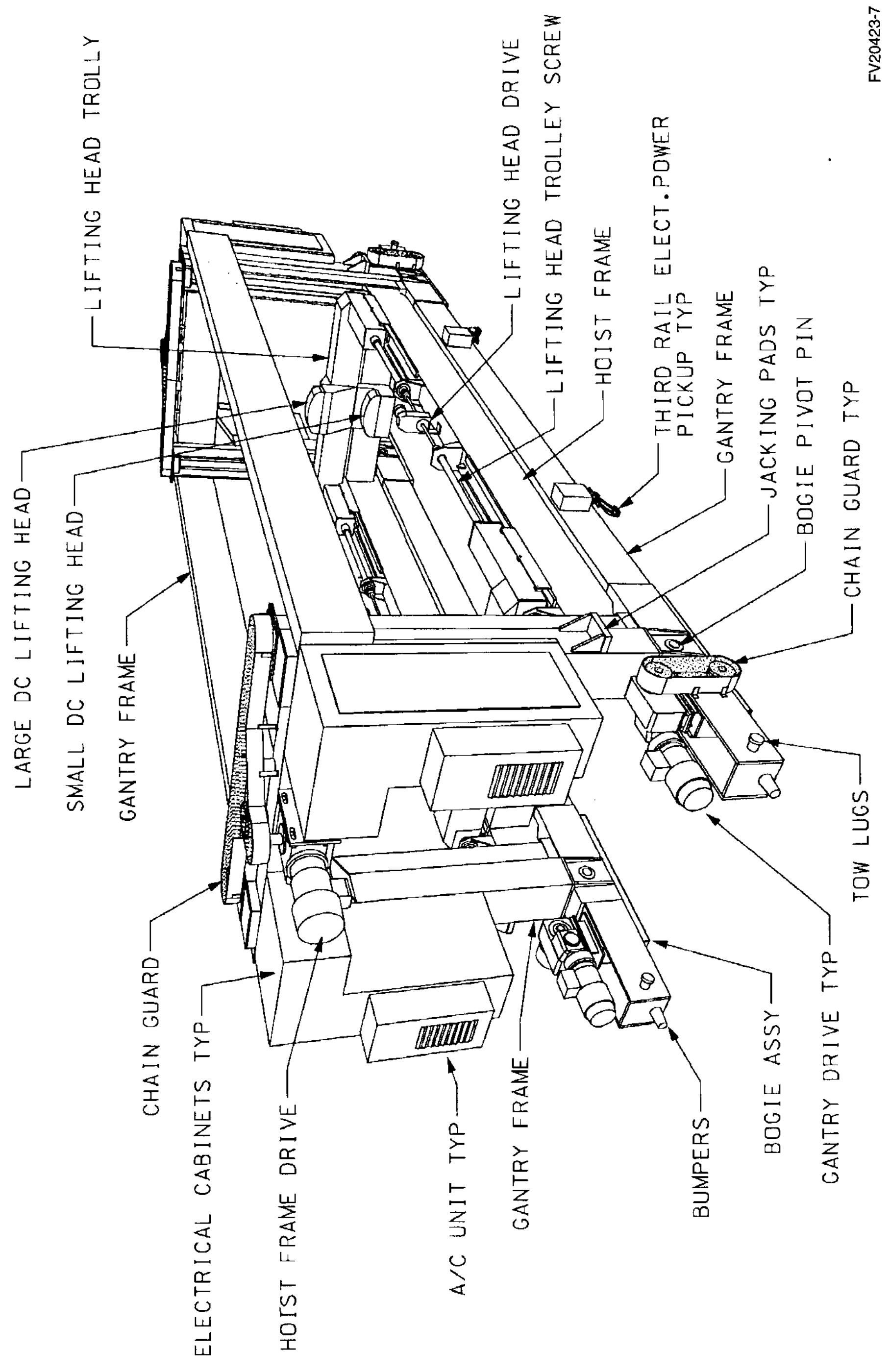

Figure 4-36. Emplacement Gantry Arrangement

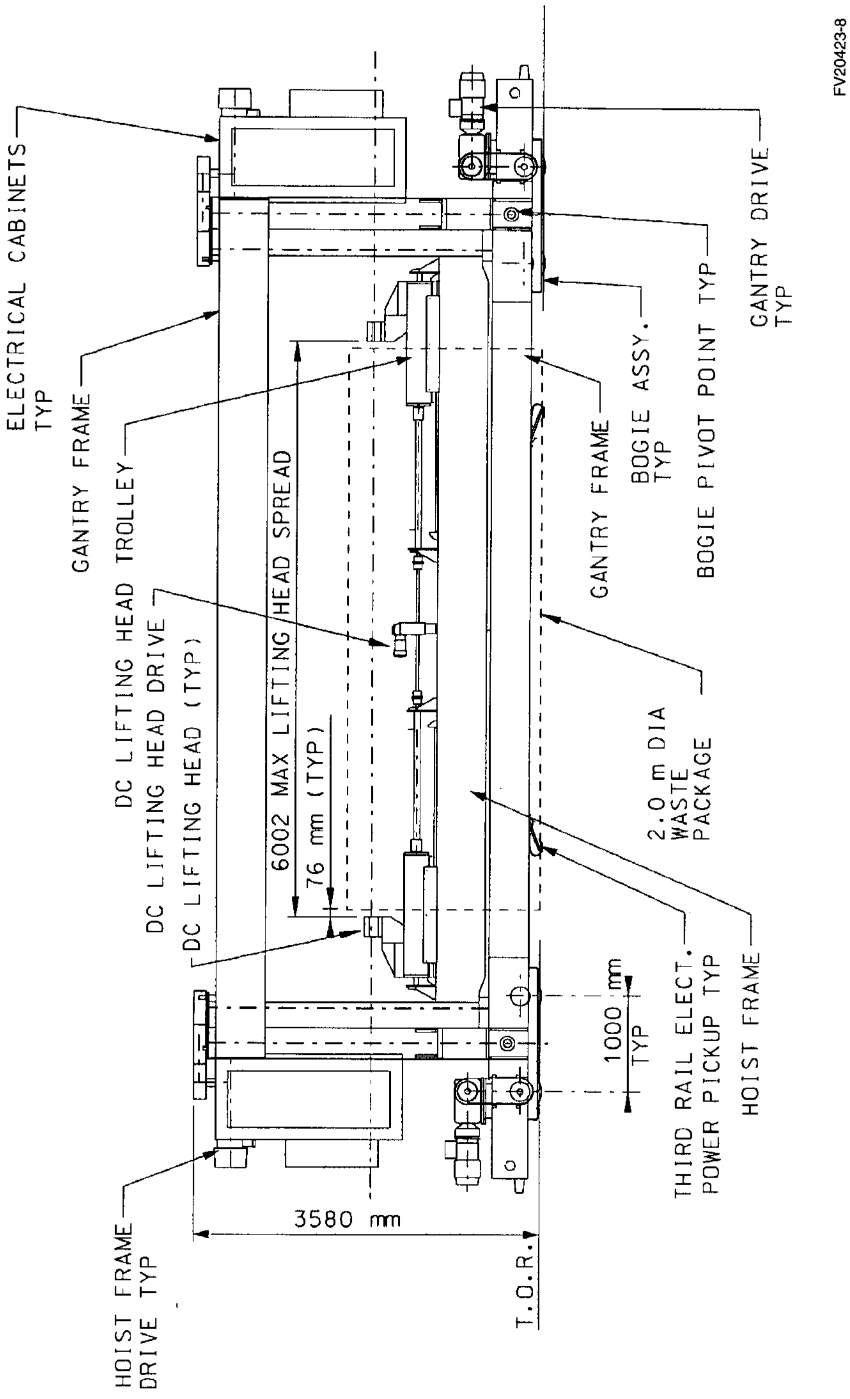

Figure 4-37. Emplacement Gantry Elevation

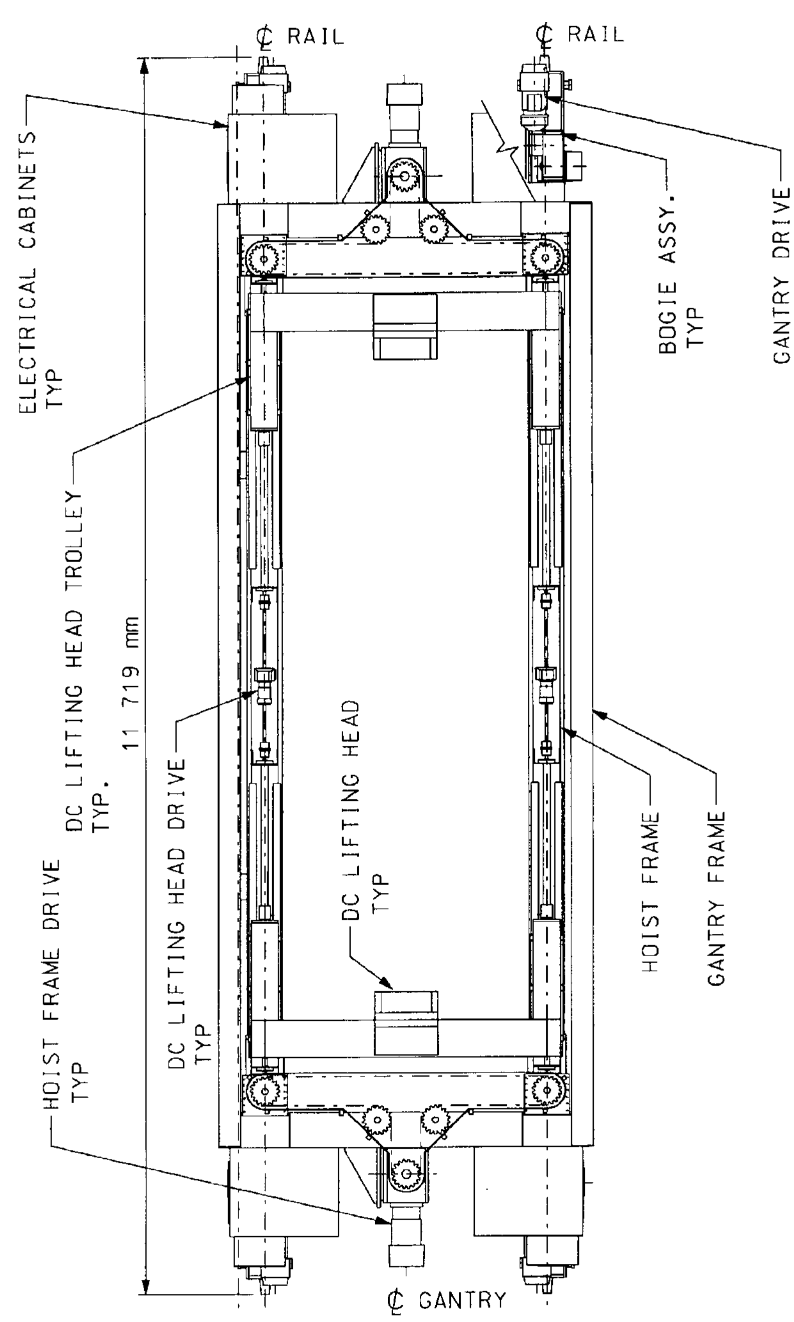

Figure 4-38. Emplacement Gantry Plan View

- Smoothly transport the waste package to the designated placement position
- Accurately position the waste package over the support assembly
- Lower the waste package onto the support assembly and disengage the lifting heads
- Return to the emplacement drift entry to receive the next waste package

The emplacement gantry is designed to operate in a 5.5-m (18-ft) diameter drift with 200 mm (7.9 in.) of radial allowance for ground support and 100 mm (3.9 in.) of radial clearance for variations in drift diameter.

The emplacement gantry will comprise several component systems including the gantry frame, traversing system, hoisting frame with adjustment capability to accommodate various waste package lengths, and lifting head trolley. Several other systems will also be required to operate and control the emplacement gantry. These systems will include closed-circuit television monitoring, braking, and systems to support the load while the emplacement gantry is traversing the drift. All operating components of the emplacement gantry will be electrically operated from an emplacement gantry power distribution system that will receive primary direct-current power from a third rail in the emplacement drift.

The emplacement gantry is designed to support its own weight, the weight of the waste package, and the loads imposed during the design seismic event. The structure will provide the bearing and drive support for the four ball-screw mechanisms that lift the waste packages and the base support for the bogies that allow the emplacement gantry to traverse the emplacement drift. The emplacement gantry structural members will be ASTM A36 steel fabricated in accordance with American Welding Society standards. ASTM A36 steel materials were selected over higher strength steels because they are durable, ductile, and resist fatigue better than the higher strength steels.

The height that a 2-m (6.5-ft) diameter waste package must be lifted during emplacement was a critical factor in determining the overall height of the emplacement gantry frame, as well as in verifying the adequacy of the 5.5-m (18-ft) drift diameter. The following two factors were evaluated to determine the lifting height:

- The first factor was the capability of the hoisting frame to lift the 2-m (6.5-ft) waste package high enough to clear the reusable rail car. A 150-mm (6-in.) clearance between the waste package and the reusable rail car structure has been incorporated into the emplacement gantry design.
- The second controlling factor was the capability of the hoisting frame to lift the waste package high enough to clear the concrete shadow shield. The concrete shadow shield will be 75 mm (3 in.) above the top of a 2-m (6.5-ft) waste package that will be placed on a support assembly. Therefore, a 75-mm (3-in.) clearance between the shield and waste package has been incorporated in the emplacement gantry design.

The height of the lift over the concrete shadow shield was the determining condition for designing the emplacement gantry lifting height. The emplacement gantry design incorporates a 2.2-m (7.3-ft) lifting height, which will provide 75 mm (3 in.) of clearance between the waste package and shadow shield. Notches on the sides of the shadow shield will provide clearance for the tow lugs. The minimum side clearance will be approximately 60 mm (2.4 in.).

The emplacement gantry will be a self-propelled unit using four, two-wheel bogies (one bogie at each corner). One wheel in each bogie will be driven by a 5-hp, direct current, variable-speed electric motor using a right-angle gear reducer and roller chain assembly. Of the alternative propulsion systems evaluated, this system will provide the greatest clearances between the drift wall and the drive mechanism. The gear motor will have an output speed of 35 rotations/minute. At this wheel speed, the emplacement gantry will be able to travel at speeds of up to 144 ft/minute; however, the travel speed will be controlled so it does not exceed 140 ft/minute. Emplacement gantry acceleration will be controlled by varying the motor voltage; a brake mounted in the motor will control speed and deceleration. A redundant restraining system will be provided to ensure that the emplacement gantry can be held in position over the waste package during lifting and placing operations, and that it remains immobilized while being transported on the gantry carrier. The use of four motors complies with American Society of Mechanical Engineers NOG-1, which requires that each four-wheel unit use a drive arrangement that provides power to at least 50 percent of the wheels. The bogie wheels will be 400 mm (15.7 in.) in diameter and have a load capacity of 19.8 metric tons (21.8 tons). Because there will be eight wheels per emplacement gantry, the total load capacity of 158.4 metric tons (174.6 tons) will be sufficient to support the weight of both the waste package and the emplacement gantry structure.

The hoisting frame, which will directly support the waste package when it is lifted by the emplacement gantry, will be designed so that a waste package can be placed horizontally in the emplacement drift with a minimum spacing of 920 mm (36.22 in.) between adjacent waste packages. The hoisting frame design incorporates the capability to adjust to varying waste package lengths. This will be accomplished with ball screws located on the outside of the longitudinal support beams. These ball screws will position the lifting head trolleys at each end of the hoisting frame to accommodate the various waste package lengths. The ball screws will be sized to withstand the acceleration load of the design seismic event and will be powered by one-third-horsepower gear motors that have an output speed of 188 rotations/minute. The speed of the lifting head trolleys will be up to 4.77 m/minute (15.7 ft/minute). The hoisting frame fabrication will be similar to that of the emplacement gantry structure, with A36 built-up structural steel shapes welded in accordance with American Welding Society standards.

Once the waste package is engaged by the lifting head trolleys, the waste package can be lifted off the reusable rail car. Lifting screws, located at each corner of the hoisting frame, will provide the vertical lift. These ball screws will be supported at both ends by rigidly mounted angular contact bearings and will be chain-driven by electric gear motors mounted on the top cross-member of the emplacement gantry. A maximum vertical lift of 2.226 m (7.304 ft) is required to lift a 2-m (6.5-ft) diameter waste package over the concrete shadow shield. This design allows the waste package to be elevated to its maximum height in approximately 3 minutes. During lifting, an integral motor brake in the traversing motors will ensure that the emplacement gantry cannot move. While the emplacement gantry traverses along the emplacement drift, solenoid-operated locks will engage the hoisting frame to support the load and minimize flexing of the ball screws.

The waste packages will be lifted by two end skirts using the two lifting heads and fixtures. The lifting head trolley will adjust to the varying waste package lengths, engage the recessed ends of the waste packages, and provide a structure for lifting the waste package. The trolley will be carried on the top flange of the longitudinal beams of the hoisting frame and positioned by trolley screws that will be carried on the hoisting frame. The trolley will move along the top flange of the longitudinal beam using flat rollers. Six rollers will carry each trolley. The trolley will be secured to the hoisting frame by sliding plates welded or bolted to the hoisting and trolley frames.

The emplacement gantry will be electrically powered. A third rail installed on the drift invert will supply power through two spring-loaded, brush-type contacts that slide along the rail as the emplacement gantry moves within the emplacement drift. These contacts will be located approximately 1 m (3.3 ft) inside of each bogie, so that if the circuit is interrupted on one contact, the other will continue to supply the gantry. Electric power from the contactors will feed into a power distribution panel on the emplacement gantry.

The control system for the emplacement gantry will use a set of redundant on-board programmable logic controllers and control computers. The control system will control and monitor vital on-board operations and functions such as the following:

- Vehicle locomotion, speed, acceleration, braking, and positioning
- Waste package hoist and lifting head drive motors with limit switches
- Load locking and latching devices

The emplacement gantry control system will also operate and interface with on-board camera and lighting systems, thermal monitoring and control systems, radiation monitoring systems, power supply and distribution systems, and remote communication systems.

The emplacement gantry controls will remotely communicate with operators who will be located at a control station on the surface. The operators will be linked to the emplacement gantry controls by a subsurface communications network. The network will consist of a fiber-optic communication line installed throughout the main and perimeter drifts. This network will provide wireless remote control within the emplacement drifts. Controls system details are described in Section 4.2.5.

The emplacement gantry design also incorporates four radiation-shielded electrical enclosures. Each enclosure will be constructed of carbon steel with a minimum thickness of 51 mm (2 in.).

Gantry Carrier. The gantry carrier will transport the emplacement gantry from the surface storage facility to the emplacement drifts and between emplacement drifts during waste placement operations. Because the combined weight of the gantry carrier and emplacement gantry will be lower than the combined weight of the waste package transporter and waste package, only one locomotive will be required to move the emplacement gantry.

The gantry carrier will be similar to a railroad flat car with 90 lb/yard rails mounted on the gantry carrier bed to accommodate the emplacement gantry. The track gauge for the rails will be 2.88 m (112.20 in.), which will match the gauge of the drift invert rail. The carrier arrangement is shown in Figure 4-39; the main components of the gantry carrier are described in the following paragraphs.

Underframe. The gantry carrier underframe structure is designed as a standard railroad flat car with horizontal main stringers running the full length of the car near its midpoint. The stringers will support the bolster plates and bolsters at each end of the car providing the connections to each undercarriage. The stringer will also provide an anchor point for the coupler located at the end of the car. Outside stringers, located at the outside edge of the car, will be attached to the main stringers with crossbeams. The outside stringers will be spaced at close intervals and run perpendicular to the main stringers. The outside stringers will be attached to the rails and support the weight of the emplacement gantry. The underframe will be fabricated of structural sections and plate with welded or bolted connections.

Rail Trucks. To provide compatibility among the various pieces of emplacement equipment, the gantry carrier rail trucks will be the same as those used on the waste package transporter. The gantry carrier will interface with the primary locomotive through a coupler provided at one end of the gantry carrier. The coupler will be the same type as that provided with the waste package transporter. The gantry carrier braking system will also be the same as the braking system installed on the waste package transporter.

Restraints. Once the emplacement gantry is loaded onto the gantry carrier, it will be clamped to a restraint designed to completely secure the emplacement gantry when the gantry carrier is in motion. The restraint will be a spring set/electric, or air pressure release fail-safe type, which engages the gantry at several locations.

Power System. A third-rail electric power system installed on the carrier will provide power for loading and unloading the emplacement gantry. This rail will be compatible with the system installed in

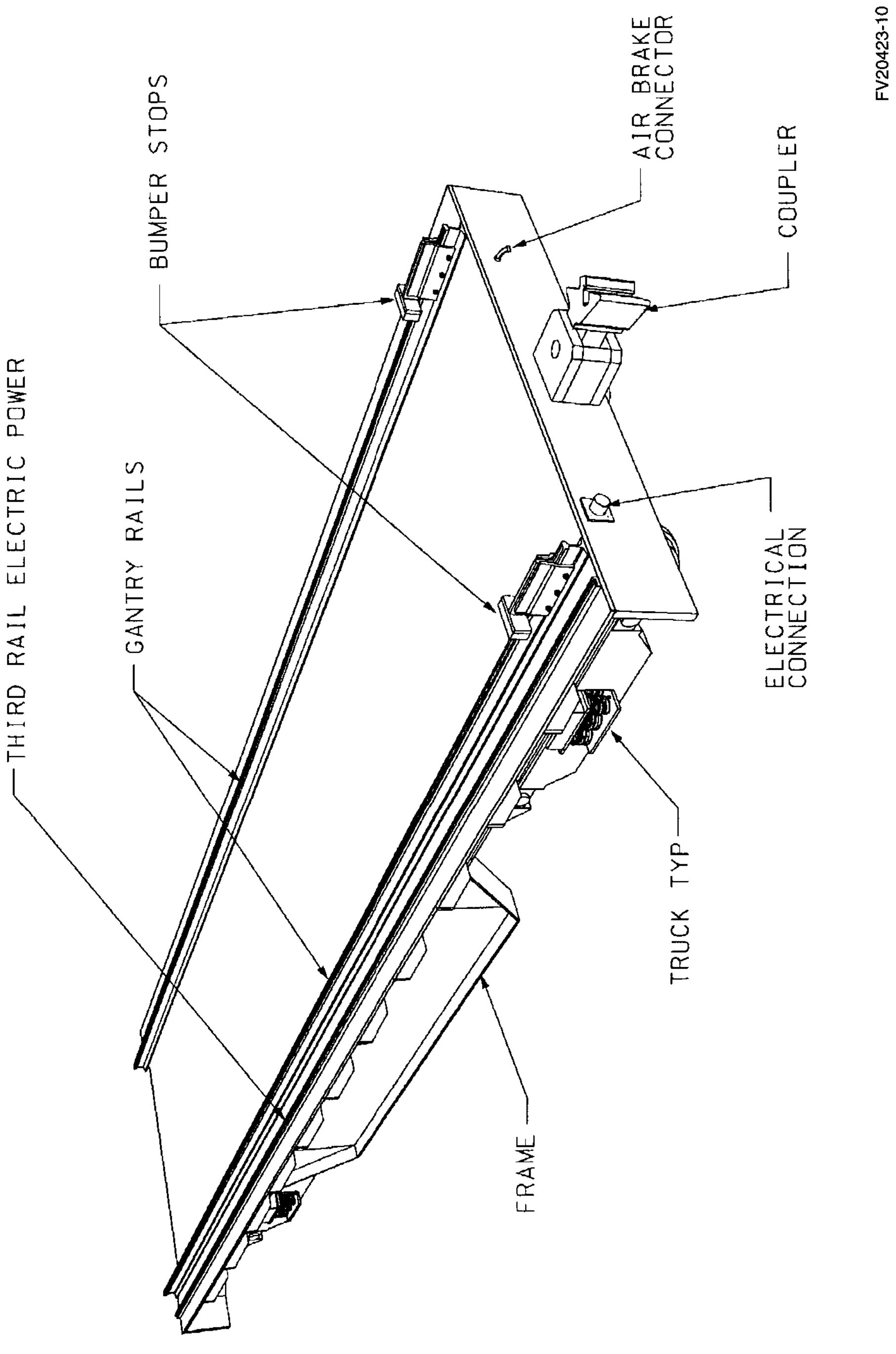

Figure 4-39. Gantry Carrier Arrangement

the emplacement drifts. Power will be supplied to the third rail from the primary locomotive through an electrical cable connection to the gantry carrier. A brush contact will be installed approximately 1 m (3.3 ft) from each end of the emplacement gantry so that power will be continually supplied to the traversing drive controls during emplacement gantry loading and unloading.

Emplacement Drift Transfer Dock and Isolation Doors. Each emplacement drift will be equipped with a transfer dock that incorporates the emplacement drift isolation doors.

Transfer Dock. The conceptual design of the emplacement drift transfer dock is illustrated on Figure 4-40. The transfer dock is designed to perform the following functions when waste packages are being transferred to the emplacement drift:

- Achieve and maintain the proper distance between the top of rail in the turnout and the top of rail in the emplacement drift to accommodate the particular transfer equipment (reusable rail car or gantry carrier).
- Allow flush contact between the edge of the transfer dock and the corresponding edge of the waste package transporter and gantry carrier.
- Accommodate any protrusion on the waste package transporter or the gantry carrier with block-outs in the dock face.
- Align, support, and maintain the alignment of the reusable rail car rail in the waste package transporter or the gantry carrier rail with the emplacement drift rail systems.
- Verify alignment and support of the transport equipment before transfer proceeds.

To unload the reusable rail car, the rails and the rigid chain drive guide will extend over the edge of the dock and line up with the respective rails and guides of the waste package transporter. At the same time, the rails and guides will be supported on the back edge of the waste package transporter floor, which will be exposed after the transporter doors open. The face of the dock will have a pocket for the coupler and will allow the dock edge to fit flush to the transporter floor. The face of the dock will also be recessed to allow clearance for the transporter door operator motors.

The gantry carrier rails used for loading and unloading the emplacement gantry will be flush with the end of the carrier and butt against the emplacement drift rails. It will not be necessary for the emplacement gantry electrified third rail to butt against the respective gantry carrier third rail, because the emplacement gantry will have two power contactors, which will allow it to always draw power from either of the two rails.

Isolation Doors. The design concept for the emplacement drift isolation doors is described in *Emplacement Drift Air Control System* (CRWMS M&O 1997i). The emplacement drift isolation doors will open and close frequently while waste packages are being placed. After waste placement is completed within an emplacement drift, the doors will be closed and will be opened infrequently. The design functions of the isolation doors include the following:

- Sealing the emplacement drift to prevent unwanted air from entering
- Regulating airflow into the emplacement drift when needed
- Providing some radiation protection for workers
- Controlling access to the emplacement drifts
- Accommodating both high and low thermal loads
- Requiring low maintenance

The design concept for the isolation doors requires that a low, controlled volume of air be allowed to enter the emplacement drift. This requirement will be achieved by equipping the door with seals that

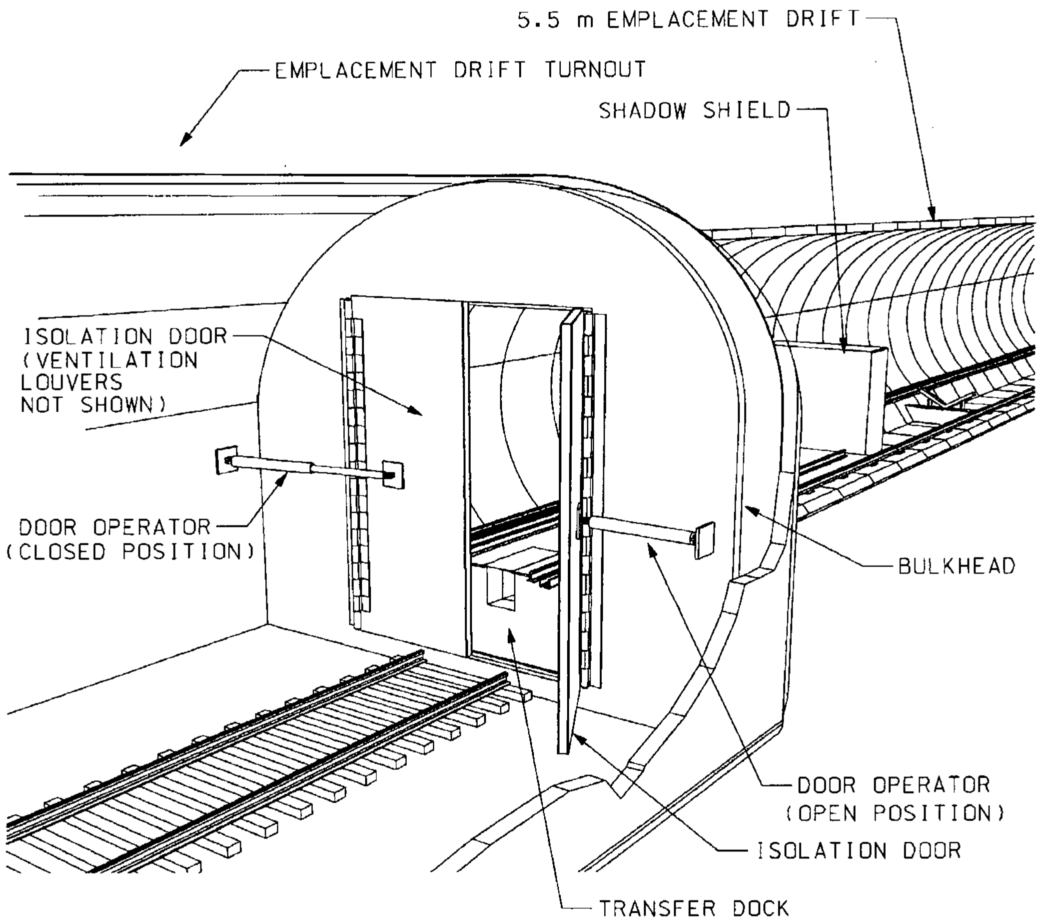

Figure 4-40. Emplacement Drift Transfer Dock

prevent or restrict the amount of unwanted air that leaks into the emplacement drift. Continuous ventilation is calculated to be no more than 0.1 m^3/s (211.9 $ft^3/minute$) (average) per drift. This quantity includes both air that leaks into the drift and air that is deliberately introduced by the louvers and dampers installed in each door. The louvers and damper will act as regulators and allow air to flow into the drift when needed. To ensure a continuous low flow of less than or equal to 0.1 m^3/s (211.9 $ft^3/minute$), instrumentation for monitoring airflow has been incorporated into the design.

The doors will be heavier and sturdier than required for structural adequacy and ventilation control to ensure that the door will provide some level of radiation protection. The door configuration will also incorporate features that prevent radiation streaming through gaps in the doors or between the doors and the door support frame.

Because the isolation doors will control access to high and very high radiation areas, a conspicuous visible or audible alarm signal will be installed at each emplacement drift entrance or high radiation area access point. This will ensure that both the individual entering the high-radiation area and the supervisor of door opening operations will be aware of entry into a high-radiation area. Entryways will be locked, except for approved access.

The temperature of the emplacement drift walls will be below 200°C (390°F), and the maximum temperature of the access mains will be 27°C (80°F). A temperature gradient through the door will occur because the door structure will be located at the interface between these two temperature zones.

The isolation doors are designed as a swing-type door system, consisting of two door panels that each pivot around a hinged point. Using two doors rather than one reduces the load on the hinged connection between the door and the supporting doorframe. The option remains, however, to use a one-door system if future analyses indicate that a two-door system allows too much air leakage. The doors will be sized to accommodate the emplacement gantry, which will be the largest piece of equipment to be loaded and unloaded at the emplacement drift transfer dock. Each of the two swing doors will be equipped with a remotely controlled actuator to open and close the door.

The doors will be constructed of ASTM Type 304 stainless steel with a minimum thickness of 1 in. This thickness will be an important factor in providing radiation protection to personnel in the main drifts when the doors are closed. The doors will be constructed of a channel frame with a Type 304 stainless steel skin on each side that will be at least 0.5 in. thick. The frame will add rigidity to the door structure, similar to a hollow-core door, and aid in retaining the door's shape under a wide range of thermal loads. Insulation will be added to the hollow door space to reduce heat transfer from the emplacement drifts to the main drifts. The door supporting frame and the drift bulkhead will complete the isolation between the emplacement drift turnout and the emplacement drift. The door and frame will be placed at the interface between the short turnout and the beginning of the emplacement drift. The doors and frame will be set in a concrete bulkhead, installed to form a seal against the rock and to fill any cracks in the rock, decreasing the amount of leakage around the bulkhead.

The louver on each door must be able to control airflow and prevent radiation from escaping the emplacement drift. The louver will control the flow of air during low-flow conditions. Ideally, no air will enter the drift when the louver is completely shut. Therefore, the only air entering the drift will be air that leaks through the door seal or through the surrounding rock mass. If airflows resulting from leakage are greater than 0.1 m^3/s (211.9 $ft^3/minute$), a cover will be placed over the louver to completely cut off airflow and to ensure that no air leaks through the louver. The volume of airflow depends on both the size of the louver and the magnitude of the pressure driving the air through the louver.

The door design calls for the permanent installation of a 16-ft^2 louver, with provisions for installing a larger louver. Modular louvers may be inserted in the emplacement doors to provide airflow as needed. The door design includes a provision that allows these modular louvers to be easily inserted. Blast cooling (rapid cooling) will be a relatively rare event, so only a few sets of the larger louvers will be needed.

The 16-ft^2 louver will be located in the door rather than the doorframe or off to one side of the emplacement drift entrance. Because the louver will be small, it can easily be located in one location, instead of being spread along the frame. The air will stratify in the drift, and the central location of the louver will tend to ensure that hot, stratified air will be kept away from the door. Air velocity will be the highest near the door; as the air proceeds down the drift, its velocity will slow to the average velocity in the drift (about 0.0073 m/s [1.4366 ft/minute]). The louver will be located near the bottom of the door in the center. This placement allows easy access for maintenance and simplifies the insertion of additional louver sections if blast cooling is required.

The requirement to prevent the unmitigated escape of airborne radiation, which could only be possible if a waste package is breached, will be partly satisfied by the pressure differential between the main drift and the emplacement drift. That is, air will always flow from the main drift to the emplacement drift. To limit radioactive particulates from escaping, a baffle or plate will be installed on the drift side of the louver.

The estimated maximum air leakage through the emplacement drift doors is 0.1 m^3/s (211.9 ft^3/minute). To control this flow rate, a competent seal will be installed to minimize the gap between the door and frame. Latching mechanisms will maintain pressure on the doors to achieve a better seal and prevent unauthorized access to the drift. The doors, seal, and louver will be tested for leakage before emplacement of waste packages begins. This test will be conducted by closing the louver and opening the gate valve at the raise to provide a suction pressure on the door.

Pneumatic cylinders will control door movement. The pneumatic operator is designed with one air cylinder on each door to provide the necessary operating force. Air will be supplied by the facility's surface-mounted compressed air system; accumulators, which store compressed air, will be used for backup. The accumulators will provide sufficient air pressure for movement in the event that the main supply line fails or the surface-mounted air compressors fail.

The pneumatic system design is the best system for achieving the required control of the emplacement drift isolation doors. An air supply will be readily available from the redundant air compressors on the surface, and the accumulators provide the necessary backup capability. The design operating pressure allows the use of standard, commercially available pipe and valves. The pneumatic cylinders, compressed air piping, and valves will be located on the positive-pressure side of the doors, so no special equipment or procedures will be required for maintenance. The pressure differential will also aid in achieving a good door seal.

4.2.3.3 Waste Transport and Emplacement Concept of Operations

Transport of waste packages to the underground repository will begin with the loading of waste packages onto the reusable rail car within the Waste Handling Building (see Section 4.1.4.2). The reusable rail car will then be loaded into the waste package transporter, and the transporter doors will be closed. The primary locomotive will then move the loaded transporter out of the Waste Handling Building. These operations will be remotely controlled through the primary locomotive controls system. Once the waste package transporter has cleared the Waste Handling Building, operators will board the locomotive and guide the waste package transporter to a position where the secondary locomotive will be coupled to the opposite end of the transporter.

The secondary locomotive is designed to provide additional power and braking capacity to the transporter as it travels down the north ramp. With this dual-locomotive arrangement, there will always be one locomotive ahead of the transporter, in either travel direction, on an incline (upward or downward) to prevent a potential runaway situation.

Two locomotives will also provide a more effective braking configuration for the descent to the emplacement drift.

The waste package transporter couplers will be designed with an automatic release. Only the rear coupler will require activation during normal operation, when the secondary locomotive will be disconnected for waste package transporter unloading. The automatic coupler on the secondary locomotive will be equipped with a remotely controlled pneumatic actuator that disconnects the locomotive from the waste package transporter. All power and control systems on the waste package transporter will be connected to the primary locomotive, which will remain coupled in normal operation. All power and control functions to be operated for the waste package transporter will be activated from the primary locomotive. The secondary locomotive will also be controlled from the primary locomotive; therefore, the secondary locomotive will not have an electric and air brake connection to the waste package transporter.

The orientation of the transporter and locomotives will be established at the rail wye at the connection to the waste handling building spur before the transporter enters the north portal air lock. Whether the primary or secondary locomotive will lead the waste package down the north ramp depends on the location of the emplacement drift where the waste package will be unloaded. The transporter will then move down the north ramp into either the east or west main drift, depending on the location selected for waste package disposal.

The waste package transporter will be moved to a position on the main drift rail adjacent to the appropriate emplacement drift. At this point, the secondary locomotive will be decoupled so that the waste package transporter will be unobstructed for docking at the emplacement drift transfer dock. Before docking, the operators will vacate the primary locomotive, and operations again will be remotely controlled. The switch for the rail turnout to the emplacement drift will be positioned to allow the primary locomotive to push the waste package transporter onto the siding rail toward the transfer dock at the emplacement drift entrance.

As the train approaches the transfer dock, the waste package transporter doors and the emplacement drift isolation doors will be opened by remote control. The waste package transporter will then be backed up to the transfer dock. Once the transporter is aligned with the emplacement drift transfer dock, the reusable rail car and waste package within the transporter will be unloaded using the transporter's internal loading/unloading mechanism.

After the reusable rail car containing the waste package is unloaded into the emplacement drift, the rail-mounted emplacement gantry in the emplacement drift will move into place over the waste package. The emplacement gantry will grip the waste package at both ends and raise it high enough to clear the end of the reusable rail car and the emplacement drift's shadow shield. The emplacement gantry will then traverse the emplacement drift to the location where the waste package will be placed on the preset support assembly. After placement, the emplacement gantry will release the waste package, and the empty emplacement gantry will return to the emplacement drift entrance to await the arrival of the next waste package. All emplacement gantry movements will be remotely controlled.

After the waste package is removed from the reusable rail car by the emplacement gantry, the reusable rail car will be retracted back into the waste package transporter. The primary locomotive and transporter containing the empty rail car will move away from the dock area. Once the transporter clears the dock area, the transporter doors and the emplacement drift isolation doors will be closed.

The locomotive and transporter will then move toward the main drift via the turnout and past the rail switch. The waste package transporter will be recoupled to the second locomotive after the turnout switch is repositioned. The operators will reboard the primary locomotive, and remote control will be discontinued. The operators will guide the locomotives and transporter through the main drift and up the north ramp back to the Waste Handling Building.

4.2.4 Subsurface Ventilation

Ventilation is a critical support function in the development of the underground repository and emplacement of the waste packages. Repository development and emplacement are distinct and separate operations, which will take place concurrently. Because of concurrent development and emplacement, each of these operations will require a separate ventilation system (see Figure 4-41). The emplacement and development operations will be ventilated by two separate and independent systems of fans and control devices. Movable underground isolation air locks will physically separate the air flows of the two systems.

Development-side ventilation will be provided by a pressure system. In a pressure system, the fans are located on the intake side, and air is pushed underground exiting the south portal. This pressure system causes the air pressure throughout the development side to be above atmospheric pressure. In contrast, the emplacement side will be ventilated by an exhaust system. In this system, the fans are located at the exhaust point and pull air in and down the north portal and through the subsurface. The exhaust system causes the air pressure throughout the emplacement side to be below atmospheric pressure.

This combination of above-atmospheric pressure on the development side and below-atmospheric pressure on the emplacement side will create a pressure differential that forces any air leakage between sides to flow from the development side to the emplacement side, where the risk of radionuclide release will exist. The net pressure differential between the development and emplacement areas may be at least 189 pascals (0.76 in. water guage). The emplacement exhaust system will be equipped with a standby high-efficiency particulate air filtration system that will come online if radionuclides are detected in the air of the exhaust main. Ventilation will be maintained in an emplacement drift until all the waste packages are emplaced in that drift. Afterward, as discussed in *Emplacement Drift Air Control System*, a low continuous flow of 0.1 m^3/s (212 ft^3/minute) is planned for each emplacement drift (CRWMS M&O 1997i, Section 8). Continuous flow-through ventilation will be maintained in the repository accesses until closure.

The combined effects of the thermal load from the waste packages and the air flow through the emplacement drifts tends to reduce relative humidity inside the emplacement drifts and remove moisture retained in the rock matrix during the preclosure period. This, in turn, serves to delay moisture-induced corrosion of the outer wall of the dual-barrier waste package thereby extending the lifetime of the waste package. This reduction in relative humidity is expected to continue until the rock begins to cool and moisture returns to the drifts.

Air flow through the emplacement drifts will be controlled at two points. The inlet end of the drift has a louvered arrangement built into the isolation door. The exhaust end will have a valve and duct arrangement. Both will be needed to regulate the air flow because the doors must be opened periodically to receive the waste packages. When the doors are open, the air flow will be controlled solely by the exhaust valve.

Two of the emplacement drifts, standby drifts, will not contain waste packages. These standby drifts will be finished with inlet and outlet structures in case the drift is needed to emplace waste packages from another drift.

Section 4.2.4.1 describes the different configurations of the ventilation system during early construction of the repository, during concurrent development of drifts and emplacement of waste packages, and during the monitoring phase of the repository. Section 4.2.4.2 describes the ventilation system components. Section 4.2.4.3 describes the behavior of the ventilation system under special circumstances: retrieval, if necessary; and backfill of emplacement drifts, if chosen as an option. Section 3.2.1.3 of Volume 4 discusses the continuing subsurface ventilation activities.

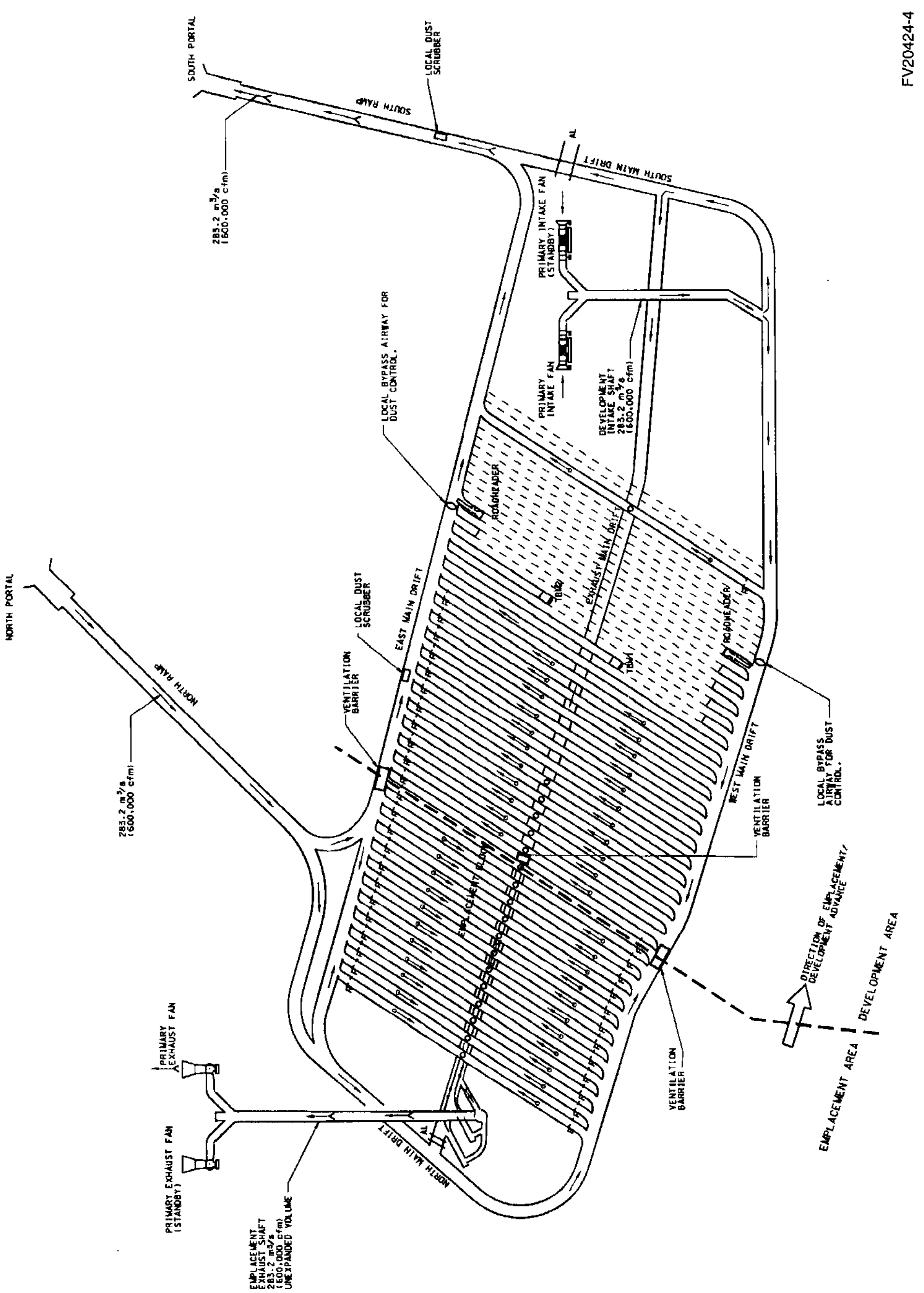

Figure 4-41. Development/Emplacement Ventilation

4.2.4.1 Ventilation Configuration During Various Repository Phases

Construction Phase Ventilation. The VA repository layout (Figure 4-42) illustrates the ventilation system for the early construction phase. The entrance, exit, and muck haulage routes will be established by repository layout and construction analyses that will provide the basis for developing the repository ventilation plan (*Repository Subsurface Layout Configuration Analysis* [CRWMS M&O 1997ab] and *Subsurface Construction and Development Analysis* [CRWMS M&O 1997af]). During initial repository construction, the low number of headings and possible air routes will limit the ventilation configuration possibilities. As construction advances and more openings become available, more efficient ventilation arrangements will be possible.

Early construction will include tunnel boring machine excavation of the south main drift, west main drift, north main drift, north ramp extension drift, east main north extension drift, cross-block drifts, exhaust main drift, and the first section of emplacement drifts and associated ventilation raises. The development intake shaft and the emplacement exhaust shaft will also be constructed during this period.

For early construction of the repository, the openings that were developed for the Exploratory Studies Facility will provide the flow path for ventilation air. Air will enter through the north ramp and exhaust through the south ramp. Auxiliary construction fans will be used to provide ventilation for the 7.62-m (25-ft) diameter tunnel boring machine during excavation of the south main drift, west main drift, north main drift, and exhaust main drift. A separate auxiliary construction fan will support the 5.5-m (18-ft) diameter tunnel boring machine during excavation of the three cross-block drifts and the connecting drifts to the development intake shaft and emplacement exhaust shaft.

Three temporary intake fans will be installed at the north portal to provide ventilation with an air lock system for equipment and personnel passage. The three fans will be activated individually as needed to support the ventilation requirements during early construction. Each fan can deliver 66.1 m^3/s (140,000 ft^3/minute) of air at 1.7 kPa (6.8 in. water gauge) static pressure. When operated simultaneously in parallel, the three fans can deliver 188.8 m^3/s (400,000 ft^3/minute) of air as discussed in *Overall Development and Emplacement Ventilation System*, (CRWMS M&O 1997q, Section 7.6.1.3).

The early construction phase ends once both emplacement exhaust shaft and development intake shaft are commissioned.

Development/Emplacement Phase Ventilation. The development phase will start after installation of the isolation barriers between the emplacement and construction sides. At that time both development intake and emplacement exhaust shaft fans will be in place and operating. When waste package emplacement operations begin, construction of the first section of emplacement drifts will have been completed. Emplacement activities in the first panel will run concurrently with development of the next section of emplacement drifts. Figure 4-41 shows the two separate ventilation systems in the emplacement and development areas during the early waste emplacement phase. Isolation barriers will be installed to separate the emplacement side from the development side. The isolation barriers will be moved southward as panels are completed and turned over to emplacement operations.

On the development side, the main fan will be located at the surface near the development intake shaft collar to force the intake air into the development network. The ventilation flow path will be from the development intake shaft fan down to the repository level; from here the air splits along the west main drift and exhaust main drift. The air will continue through the emplacement drift construction area and exhaust through the east main drift to the south ramp and then to the surface at the south portal pad.

Flow-through ventilation will use available drifts as airways. Because drift airways have minimum resistance to airflow, ventilation operations not

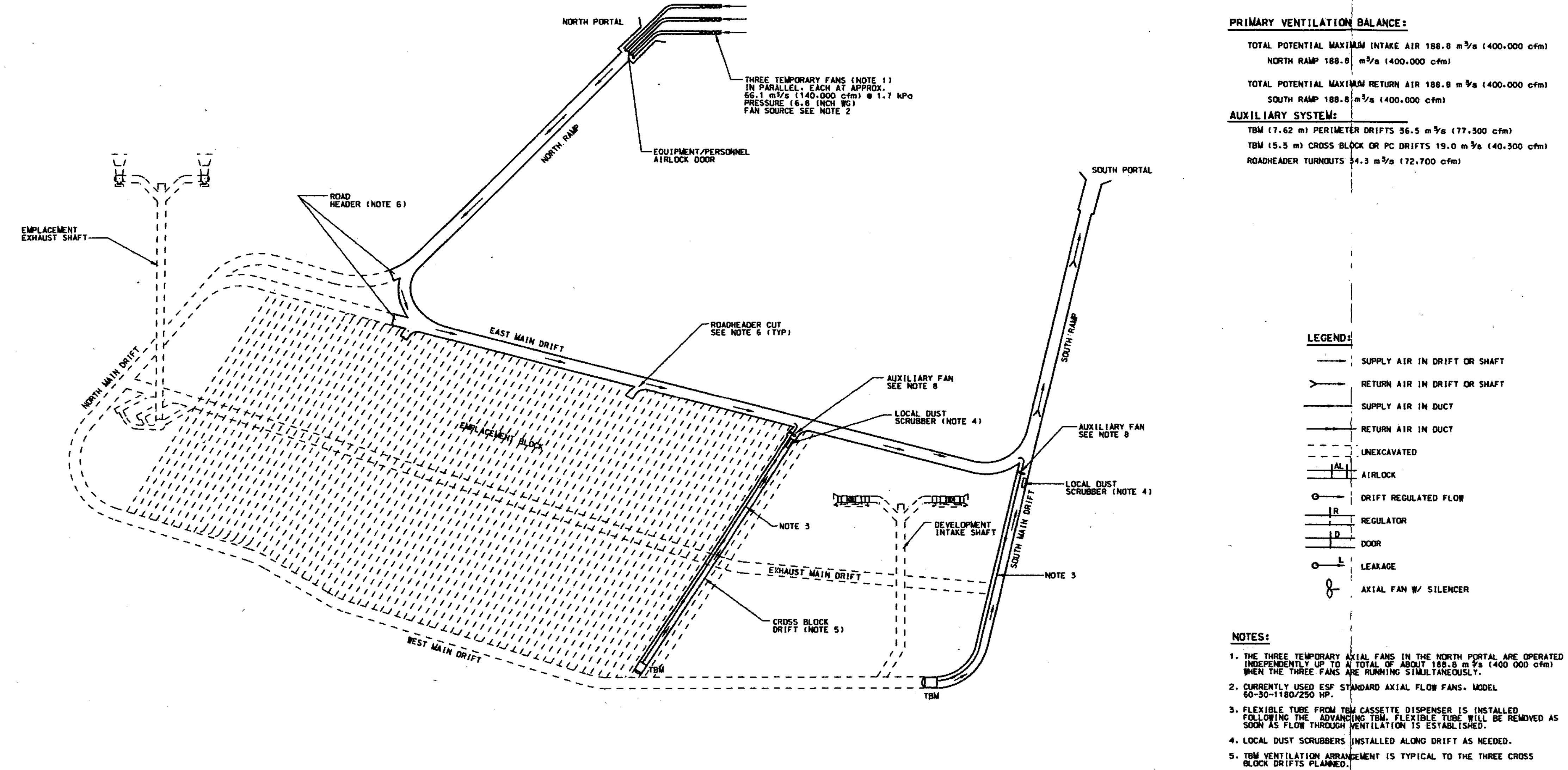

Figure 4-42. Phase 1 Early Construction Ventilation

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only permit rapid increase of airflow to area in need, but also cost less. The tunnel boring machine and roadheader will be supplied with air by an auxiliary construction fan and duct system. Other options will be investigated for the development side that use more extensive ducting and may provide better dust control. One option would have ducting coming from the surface, as a supply, to the excavation areas, and an exhaust duct running from the excavation areas back to the surface. The air balance would be adjusted to allow free air for general ventilation to enter down the ramp. The other option under investigation has the intake air being supplied down the drifts and the exhaust air being captured in ducts and routed back to the surface.

On the emplacement side, the north ramp will serve as the primary intake airway for the emplacement operations area. During repository operations, there will be no potential dust sources in the north ramp except ambient surface dust and pollens. Surface dust and other particulates can be limited by filtering the intake air through a coarse filtration system at the north portal.

The emplacement-side exhaust shaft at the north end of the repository block will serve as the exhaust airway for waste emplacement operations.

The emplacement-side fans will be located on the surface near the exhaust shaft collar. Two identical fans will be provided, one operating and one standby. Each fan will be capable of providing the capacity listed in Section 4.2.4.2.

The emplacement intake air will enter the facility through the north ramp to the east main drift and split into the following two air flows: one will ventilate the east side emplacement drifts and the second will direct air around the north end of the repository into the west main drift. Air in each of the mains will be distributed to the individual emplacement drifts and then exhausted to the exhaust shaft through the exhaust main.

The ventilation system will include two 1.829-m (72-in.) diameter vent ducts located in the exhaust main to isolate air coming from the emplacement drifts. Each of the two vent ducts will carry 47 m^3/s (100,000 $ft^3/minute$) of air; the remaining approximately 189 m^3/s (400,000 $ft^3/minute$) of air will go directly to the exhaust main drift. The vent ducts will be connected to the emplacement drifts through the control dampers located at the raise.

If a waste package breached in one of the emplacement drifts, sensors at the discharge end of the drift would initiate a response program that would contain and collect the contaminants. The high-efficiency particulate air filter units located at the exhaust shaft would be shifted from an offline mode to online. This would be accomplished by a preprogrammed response where the exhaust fans for the duct system will be slowed to reduce the air flow from 47 m^3/s (100,000 $ft^3/minute$) to 14.2 m^3/s (30,000 $ft^3/minute$) to match the capacity of the high-efficiency particulate air filters. Control dampers in the duct system will be programmed to activate in a manner not to exceed the high-efficiency particulate filter capacity. Isolation and control valves in the duct system will be placed so that air flowing in either duct can be routed to either set of filters or fan.

Monitor Phase Ventilation. The monitor phase will begin after the final waste package has been emplaced. The purpose of the ventilation system during the monitor phase is to support subsurface personnel in the maintenance of airways, environmental monitoring and performance confirmation, and keeping the repository ready for retrieval, backfill, and closure. Figure 4-43 illustrates the ventilation system during the monitor phase.

The emplacement-side shaft will be used as the exhaust airway in the final emplacement phase. During the monitor phase, the emplacement-side exhaust shaft will continue to be used as the exhaust airway. Since the development phase of the repository will be complete, there will be no need for the development-side system. The development-side system will be idle and may be converted to a backup exhaust shaft for the subsurface emplacement area.

During the monitor phase, the air will enter through the north ramp and/or the south ramp and

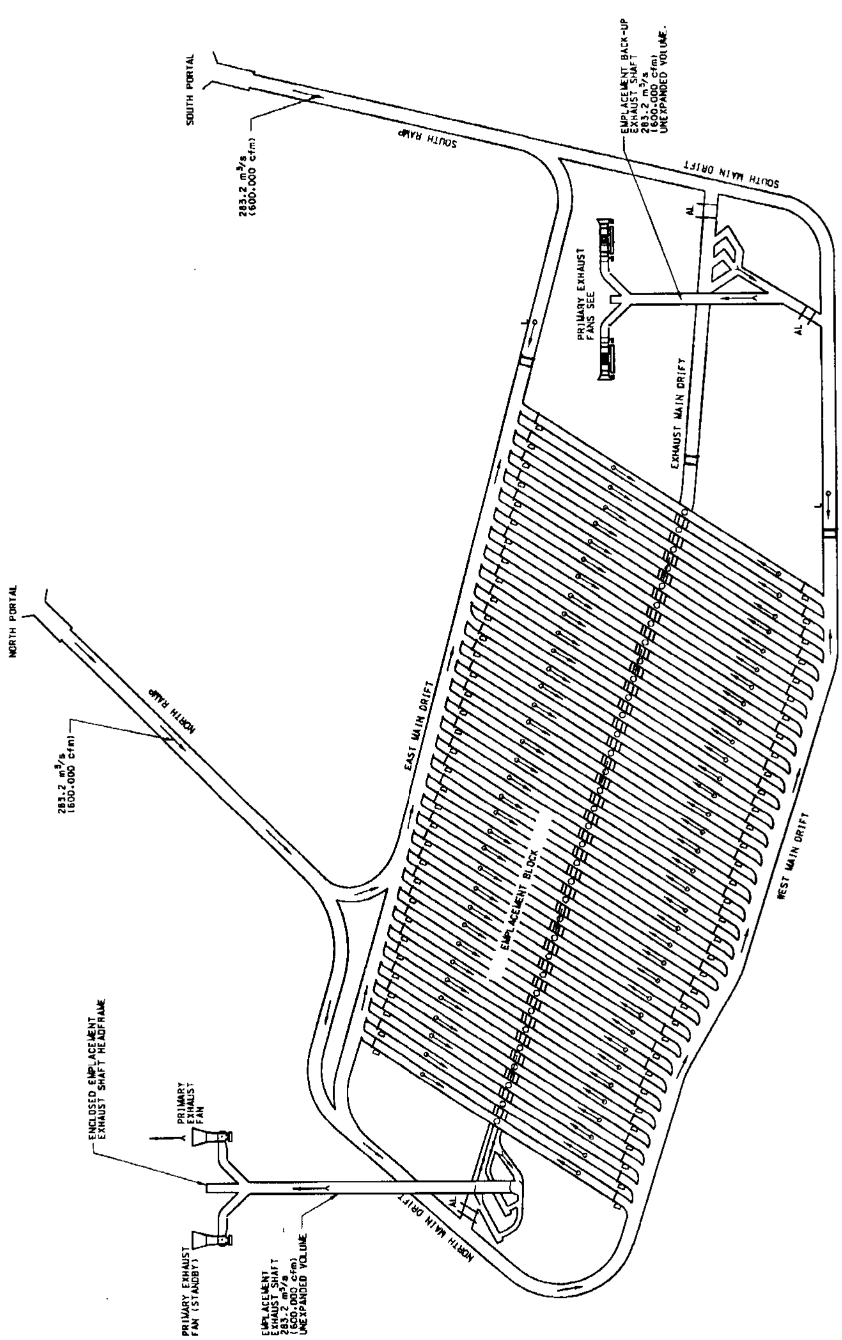

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Figure 4-43. Monitor Phase Ventilation

be split into the east and west mains. From there, the air will travel through the emplacement drifts down the ventilation raise, and into the exhaust main or exhaust main auxiliary ducts, which will be equipped with a high-efficiency particulate air filter system. Both of the routes will lead to the emplacement-side shaft or the development-side shaft, if the latter has been converted to a backup exhaust shaft and exhausted out of the tunnel.

4.2.4.2 Ventilation System Components

Emplacement-Side Fans. Two fans (one main fan and one backup fan) will be located on the surface, on a ridge north of Yucca Crest at the emplacement-side ventilation shaft. The emplacement-side ventilation fan will pull air from the north portal through the emplacement side of the repository. An emergency headframe will be located over the shaft, requiring that a transition structure be installed between the fan and surface collar.

A surface power line and backup power generator will supply power to the fans and the escape hoist. The backup fan, with performance capabilities similar to the main fan, will be installed next to the main fan.

Standard industry fans will be used, although special controls and monitoring sensors may be needed for repository purposes. The main and backup emplacement-side fans will have variable speed motors and will meet the following performance specifications:

- Maximum volume:
  360 m^3/s
  770,000 ft^3/minute

- Maximum pressure:
  3.09 kPa
  12.4 in. water gauge

- Maximum power:
  1500 kW
  2,000 hp $_{(Brake)}$

- Minimum volume:
  167 m^3/s
  353,000 ft^3/minute

- Minimum pressure:
  0.990 kPa
  3.98 in. water gauge

- Minimum power:
  165 kW
  221 hp $_{(Air)}$

The fans will be equipped with a monitoring package to gather and transmit operating data to the surface facility at the north portal. The primary data to be monitored will include the following:

- Vibration
- Bearing temperatures
- Motor current
- Fan pressure

The fans will be equipped with silencers to reduce noise to below 85 DbA at 3 m (9.8 ft) from the fan. The fans will also be provided with exhaust stack outlets and diffusers.

Development-Side Fans. Two development fans (one main fan and one backup fan) will be located on the surface, at the south end of Yucca Crest at the development shaft collar. The development-side fan will force air into the repository construction areas through the development shaft. After ventilating the construction work areas, the air will exit the repository at the south portal. A backup fan will be installed adjacent to the primary fan and both will be connected to the shaft through a transition structure. An emergency hoist and an enclosed headframe will also be located at the shaft.

A standard mine fan can be used for the repository construction and development. The fan will be provided with a backup power source and will include a reversible starter switch. The main fan and the backup fan will have variable speed motors and will meet the following performance specifications:

- Maximum volume:
  300 m^3/s
  635,000 ft^3/minute

- Maximum pressure:
  3.7 kPa
  15.0 in. water gauge

- Maximum power:
  1500 kW
  2,000 hp $_{(Brake)}$

- Minimum volume:
  178 m^3/s
  376,000 ft^3/minute

- Minimum pressure:
  0.687 kPa
  2.76 in. water gauge

- Minimum power:
  122 kW
  164 hp $_{(Air)}$

The surface fan installation resembles the configuration at several underground mining operations. Silencers will be installed to reduce noise levels to less than 85 DbA at 3 m (9.8 ft) from the fan. Fan operating data will be monitored and transmitted to the south portal operations center. The data to be monitored will include the following:

- Vibration
- Bearing temperatures
- Motor current
- Fan pressure

Auxiliary Construction Fans. Auxiliary construction fans are needed for the tunnel boring machines and roadheader excavations. The auxiliary fans will deliver clean air to the tunnel boring machine faces through ventilation ducting. The auxiliary fans will remove dust-contaminated air from the source through ventilation ducts to a dust collector/scrubber system, then discharge the air into the main exhaust airstream, where it will be diluted, and exits at the south portal. The control of silica dust is not discussed in the VA reference design. The auxiliary fans will be standard mine fans commonly used in underground tunneling operations. All auxiliary fans will be equipped with silencers to reduce noise levels to less than 85 DbA at 3 m (9.8 ft) from the fan.

Auxiliary fans will operate at 480 volts with operating pressures that can be varied by adjusting the fan's blade settings. Fans may either push air through rigid or flexible vent tubing, or exhaust air from rigid vent tubing. Data monitoring for all auxiliary fans will be considered, although fans on the tunnel boring machines may require that performance data be transmitted to the surface.

Construction Ventilation Duct. Ventilation ducts are required to remove dust-laden exhaust air from some excavation areas and to provide clean air to others. Flexible ventilation duct will be supplied in specially designed cassettes so that the duct can be quickly installed behind the tunnel boring machines. The preloaded cassettes will dispense 300-m (984-ft) lengths of ventilation duct for the 7.62-m (25-ft) diameter tunnel boring machines and 150-m (492-ft) lengths of ventilation duct for the 5.5-m (18-ft) tunnel boring machines. Flexible duct is generally fabricated from fire resistant plastic and fiber materials.

A 1.83-m (6-ft) diameter flexible duct is considered adequate to support the 7.62-m (25-ft) diameter tunnel boring machine excavation of the main drifts and the exhaust main. The 5.5-m (18-ft) diameter emplacement drift tunnel boring machine will be supported with a 1.37-m (4.5-ft) diameter flexible duct.

Some excavation operations may require rigid ventilation duct constructed of fiberglass or rolled steel. This duct is considered a temporary item and will be removed after the development construction activities have ended.

Dust Collectors and Scrubber Systems. Dust collectors and scrubber systems clean the dust-laden air generated by excavation, drilling, and muck handling operations.

Dust collectors are self-cleaning units, equipped with low noise in-line centrifugal fans, motor starters, and dust disposal containers. With current technology, efficiencies of 90 percent dust removal down to 1.0-micron airborne particulates are available. Dust collectors are available as stand-alone portable units or as mobile units mounted on flat cars. Some mobile and portable dust collectors are specified as dry units to maintain versatility for subsurface applications where water use and disposal could be a factor.

Dust scrubbers will be placed at strategic locations in primary airways and in selected drifts, and will include both mobile and portable units. All tunnel boring machines will be equipped with wet dust scrubbers to limit the amount of airborne dust both at its primary source and during material rehandling.

Emplacement Drift Isolation Doors. Access to the emplacement drifts from the perimeter mains will be through remotely operated emplacement drift isolation doors. The emplacement drift isolations door will open and close frequently while waste packages are emplaced. After waste package emplacement is completed within an emplacement drift, the doors will be closed and open only infrequently. The exit from the emplacement drift raise to the exhaust main will be through an air ducting structure, which will include valves and an access port.

The emplacement drift isolation doors will be equipped with louvers to regulate the air flow through the emplacement drift on the emplacement side of the repository. These doors are designed as permanent fixtures to provide ventilation control and reduce gamma and neutron radiation emissions from the emplacement drifts to the main drifts. The steel isolation doors can be fabricated on the surface for installation underground. The doors will be equipped with the instrumentation necessary to detect, and transmit to the control center, pressure differential, radiation, and open and closed status.

An exit structure with valves will be installed on the exhaust side of the emplacement drifts. These fabricated steel structures will be located at the bottom of the exhaust raises. Environmental instrumentation will be installed near the exit structures to monitor radiation, wet and dry bulb temperature, and air flow.

Emplacement-Side Exhaust System and High-Efficiency Particulate Air Filters. The emplacement-side exhaust system will include high-efficiency particulate air filters and adsorption units at the emplacement-side shaft that will come on line if a radiological release is detected. During normal operations, the filters will be bypassed. The system will be equipped with sensors and monitors that transmit data on fan performance and air quality to the repository operations center.

The subsurface portion of the emplacement-side exhaust system will be installed in the exhaust main with the fans for the system located near the bottom of the emplacement-side shaft (Figure 4-44). The system will include two 1.83-m (6-ft) diameter steel ducts in the exhaust main fabricated from 6-mm (0.24-in.) steel plate (Figure 4-45). The ducts will be connected to each emplacement drift via steel structures and valves. The system will control the airflow in the emplacement drifts and will, if necessary, cool the emplacement drift for retrieval. The ventilation ducts will prevent the spread of contamination in the event of a radiological release in an emplacement drift due to a breached waste packages.

The fans for the subsurface emplacement-side exhaust system will be standard industry centrifugal fans provided with backup power. The fans will be equipped with silencers that reduce noise to below 85 DbA at 3 m (9.8 ft) from the fan. The fans will have variable speed motors and will meet the following performance specifications:

- Maximum volume:
  64.7 m^3/s
  137,000 ft^3/minute

- Maximum pressure:
  11 kPa
  43 in. water gauge

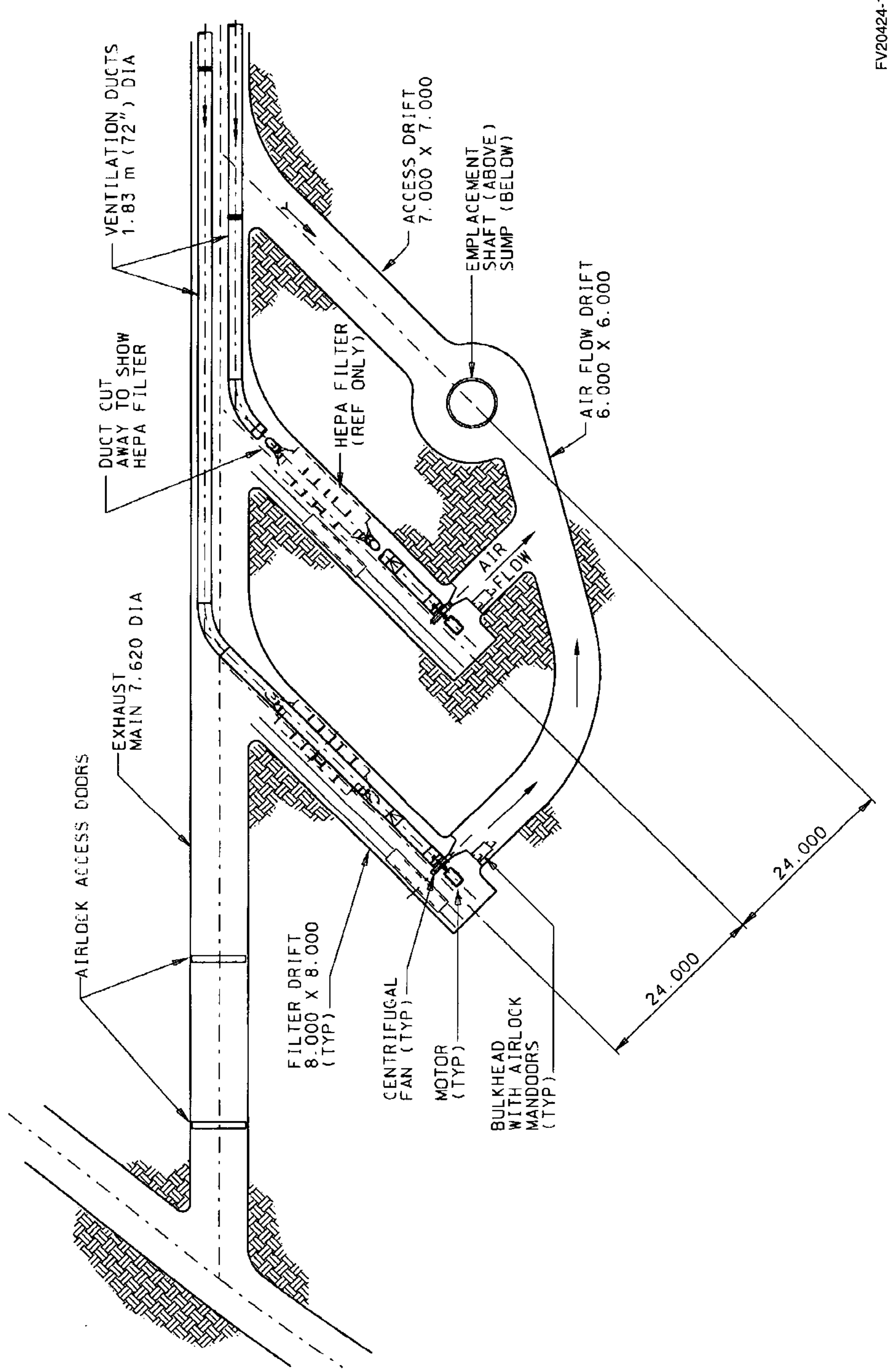
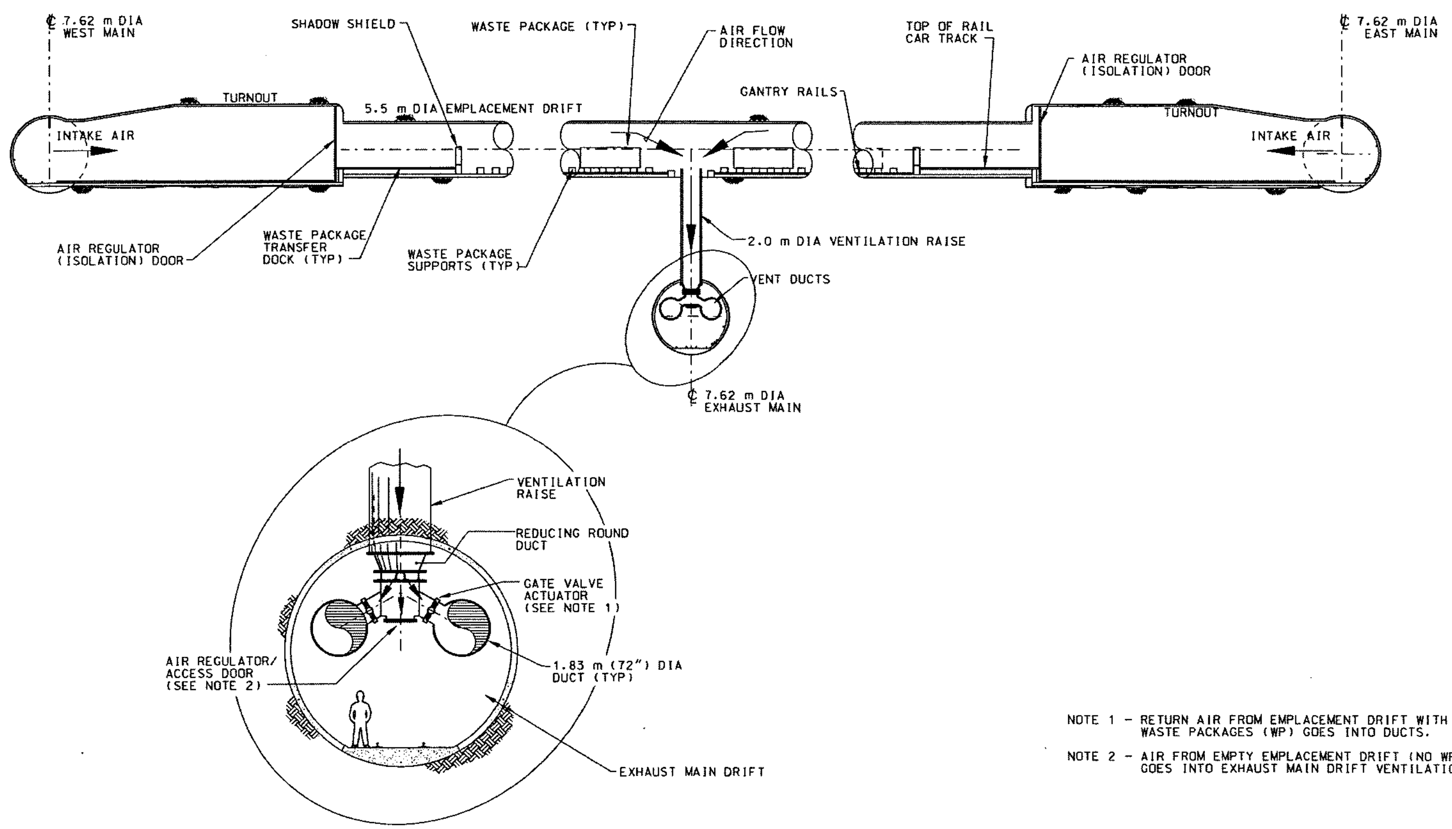


Figure 4-44. Drift Arrangement for Auxiliary Emplacement Exhaust Fans



NOTE 1 – RETURN AIR FROM EMPLACEMENT DRIFT WITH WASTE PACKAGES (WP) GOES INTO DUCTS.

NOTE 2 – AIR FROM EMPTY EMPLACEMENT DRIFT (NO WPs) GOES INTO EXHAUST MAIN DRIFT VENTILATION.

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Figure 4-45. Emplacement Drift Ventilation

- Maximum power:
  932 kW
  1,250 hp (Brake)

- Minimum volume:
  48.6 m^3/s
  103,000 ft^3/minute

- Minimum pressure:
  1.42 kPa
  5.7 in. water gauge

- Minimum power:
  69 kW
  93 hp (Air)

Portable high-efficiency particulate air filter systems mounted on rail cars will be available to clean the air coming from a contaminated area. These portable filters could be used to assist in decontaminating an emplacement drift or one of the main drifts if a radiological release occurred.

Isolation Barriers. The development side and the emplacement side of the repository will be separated by isolation barriers installed in the east main, west main, and exhaust main. These barriers will isolate the emplacement-side and development-side ventilation air streams. The barriers may be erected at several locations and be relocated periodically until repository development is completed. The pressure difference across the barriers will ensure that any air leakage will flow from the development side of the repository to the emplacement side.

Human factors can affect the isolation barriers in the following three ways:

- The barriers can be accidentally or deliberately damaged.

- The barrier seal can be broken if personnel pass through the manway.

- If personnel are injured, they may need to be evacuated through any one of the isolation barriers.

A single isolation barrier will not meet its functional requirements if personnel must pass through the barrier. A dual- or multiple-barrier system with an air lock will allow passage between the development and emplacement sides without breaking the barrier seal. The manway opening in each barrier will contain a self-closing door. By placing the air lock doors so that the one on the development side swings outward and the one on the emplacement side swings inward, the doors will tend to be self-closing because of positive air pressure from the development side and negative air pressure from the emplacement side.

The dual isolation barriers would typically be installed in straight, consistently round sections of the main drifts, offset from emplacement drift turnouts, curves, and cutouts. Dual barriers forming air locks will be placed to accommodate these inconsistencies in the drift cross section, while single barriers may be placed somewhat randomly.

The isolation barriers will be constructed of steel plate with a steel, wide-flange frame. The frame will be anchored to the cast-in-place concrete lining. The barriers are designed to allow personnel passage from one side to the other in the event of an emergency. The rail gauge at the development side of the barriers will not be the same as the rail gauge on the emplacement side. Rail traffic could pass from one side to the other in an emergency using special means. Emergency rail traffic would be permitted through the barrier by removing panels from the isolation barriers.

The isolation barriers will be equipped with sensors to monitor the pressure difference between the emplacement side and the development side. Sensors will also monitor the status of the manway to detect any individuals who pass from one side to the other. The isolation barriers will be temporary structures and will be moved as the emplacement side of the repository advances. The isolation barriers are designed so that they will be cost effective and easily assembled and disassembled.

Emplacement Drift Air Control System. Air will flow from both the east and west mains down each emplacement drift to the central raise. The air will then flow down the raise to the exhaust main where it will be drawn to a shaft and discharged from the repository. The repository configuration will change as different phases will be completed, so ventilation requirements for the emplacement area will change. However, the overall flow pattern for each emplacement drift will not change.

Redundant inlet and outlet structures are required so that ventilation can still be controlled when one structure is open or undergoing repair. In addition, capability will be provided to close off the entire emplacement drift with inlet and outlet controls in the event of a breached waste package. If a breached waste package were detected, the emplacement drift could be closed off by closing the exit structures in the vent raise and by closing the louvers in the isolation doors at both ends of the drift. Figure 4-45 illustrates emplacement drift ventilation discharge into the ventilation ducts in the exhaust main.

4.2.4.3 Ventilation Under Special Circumstances

Retrieval. Retrieval of one or all of the waste packages from an emplacement drift may be required during the preclosure phase. Retrieval will begin by blast cooling the drift with air to lower the temperature to 50°C (122°F). The *Retrievability Strategy Report* (CRWMS M&O 1997ad) recommends that the air temperature be lowered from 200°C to 50°C (392°F to 122°F) within 30 days. The capacity of one duct would be available for blast cooling. The other duct would collect the continuous ventilation air and any air used for monitoring. Using this system, waste packages could be retrieved from only one emplacement drift at a time.

Backfill. The option of backfilling the emplacement drifts will be considered (see Section 5.3.1). If the backfilling option is selected, the drift temperature will have to be blast-cooled to 50°C (122°F) before backfilling. During backfill operations, air will be discharged through the access port in the exit structure. The discharged air will be pulled through a filter to remove any dust generated by the backfill operation and then be vented into the exhaust main. The emplacement drifts would be backfilled to cover the waste package, leaving the top of the emplacement drift open to maintain air circulation. The procedure will allow only one drift to be backfilled at a time.

The limited capacity of the two ventilation exhaust ducts will limit the total capacity of the subsurface ventilation system. Because of this limitation, neither backfill, recovery, nor retrieval can occur simultaneously with emplacement operations.

4.2.5 Subsurface Repository Monitoring and Control Systems

The subsurface repository design encompasses a variety of monitoring, control, and data communication systems. As illustrated in Figure 4-46, subsurface waste emplacement and performance confirmation activities, as well as possible waste retrieval operations, involve an extensive network of instrumentation and digitally based monitoring and control systems. This section outlines the initial monitoring and control concepts developed for several key subsurface systems as part of the VA design effort.

Initial work has focused on developing preliminary designs of control systems for key elements of the subsurface repository. The priority has been to develop control systems for systems, structures, and components that are unique to the repository, that may be important to radiological safety, and for which there is limited regulatory precedent.

Section 4.2.5.1 describes control system concepts for the transport locomotive, waste package transporter, emplacement gantry, emplacement drift isolation doors, and rail switches (these waste emplacement components are described in detail in Section 4.2.3). Section 4.2.5.2 describes the subsurface data communication network for emplacement systems. Section 4.2.5.3 describes the elements of the performance confirmation data acquisition system. Additional subsurface monitoring and control systems are described in Section 4.2.5.4.

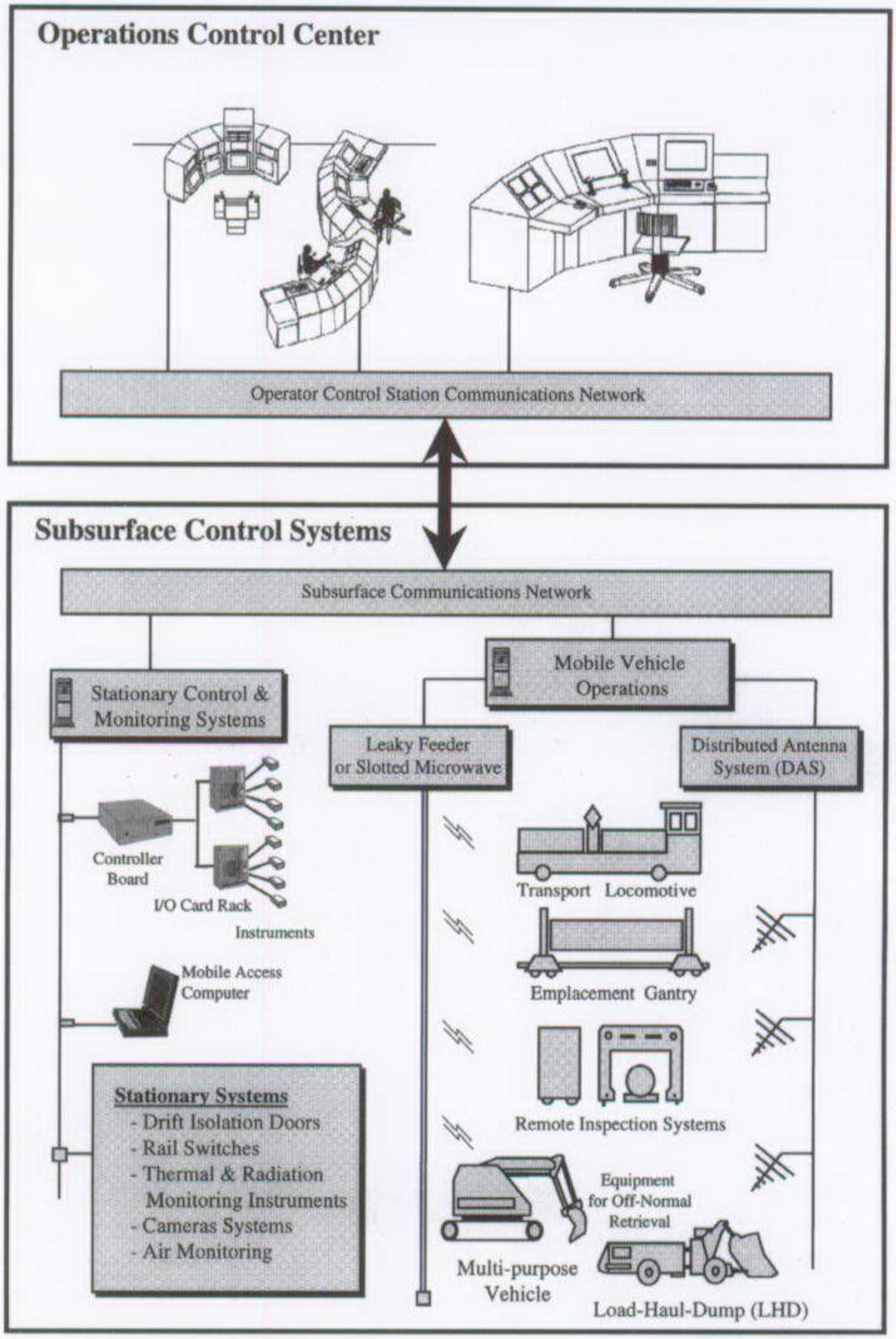

Figure 4-46. Subsurface Repository Control System

Control components for the subsurface ventilation system are described in Section 4.2.4; additional details are provided in *Overall Development and Emplacement Ventilation System* (CRWMS M&O 1997q). The waste retrieval system is addressed in Section 4.2.7, with additional detail provided in *Waste Package Retrieval Equipment* (CRWMS M&O 1997ar).

4.2.5.1 Waste Emplacement Monitoring and Control Systems

Section 4.2.3 describes the waste emplacement process, which involves multiple rail-based heavy-haul vehicles, several remotely actuated loading and unloading mechanisms, and a series of remotely activated rail switches and emplacement drift isolation doors. This section outlines the control strategy and design concepts for these key waste emplacement systems.

Transport Locomotive Control Systems. The primary and secondary transport locomotives are designed for both direct manual and wireless remote control. The locomotives will be manually controlled by on-board operators during transportation of waste packages from the Waste Handling Building on the surface to a subsurface area near the entrance of the assigned emplacement drift. The locomotives will be remotely operated when conditions are unsafe for human operators, such as during operations near the emplacement drift transfer dock.

The basic locomotive design is based on commercially available mining locomotives equipped with conventional manual controls including throttle and brake controls, as well as a variety of on-board performance gauges and indicators. In addition, the locomotives will be outfitted with commercially available equipment that provides for fail-safe wireless remote control and operation. Use of redundant, high-quality components and backup systems will ensure high system reliability.

When the locomotives are operated manually, supervisory operators located at a remote command center will continuously monitor the locomotives to ensure that they are operated properly and safely. During operation near the emplacement drift transfer dock, the locomotives will be operated entirely by operators at the remote command center. Articulated cameras and lighting systems and on-board microphones will provide operators at the command center with real-time, high-resolution visual and audio feedback. Camera systems will be mounted both at the front and back of each locomotive, as well as in the emplacement drift turnout. The operators at the command center will also receive continuous feedback on locomotive performance parameters including speed, acceleration, braking, power, communication system status, and equipment operating temperatures. The operators at the command center will have emergency shutdown capabilities.

The initial locomotive control system design is based on a dual-redundant, fault-tolerant programmable logic controller-based control configuration. The redundant programmable logic controllers will employ an input/output "voting strategy" that requires both programmable logic controllers to agree before implementing a command or providing system feedback. Additional levels of redundancy and reliability can be provided by properly programming software systems and by implementing design strategies such as using diverse technologies, physically separating redundant components, and providing backup power and communication systems.

During transit, the waste package transporter will be positioned between the two locomotives, which will be electrically and pneumatically coupled across the transporter to work in unison. By configuring the appropriate switch settings, one locomotive is designated as the lead, or primary locomotive, and the other is designated as the secondary locomotive. When working in tandem, only the primary locomotive will be manned. Each locomotive will be provided with the hardware and software necessary to operate in either the primary or secondary role.

As discussed in *Repository Rail Electrification Analysis*, a 600-volt direct-current overhead trolley wire system will provide primary power to the transport locomotives (CRWMS M&O 1997z).

Two independent and technologically diverse digital communication systems will be used to communicate with the transport locomotives. Leading candidate communication technologies will include a distributed antenna system and either a radio-frequency coaxial cable system or slotted microwave wave guide technology as indicated in *Subsurface Waste Package Handling - Remote Control and Data Communications Analysis* (CRWMS M&O 1997ai). The radio-frequency coaxial cable system is referred to as "leaky feeder" technology and is currently used in the mining industry. The slotted waveguide technology is currently used in rapid transit and rail-based people-mover industries. Additional details on the transport locomotive monitoring and control systems are provided in *Emplacement System Control and Communication Analysis* (CRWMS M&O 1997j, pp. 45–53).

Waste Package Transporter Control System. Because of the high radiation levels associated with the waste packages, waste package loading and unloading operations will be remotely controlled. In the remotely controlled mode, the operators at the remote control console will initiate loading or unloading control commands when feedback signals indicate that the transporter is properly positioned. Communication and control signals will first be processed by the programmable logic controller on the primary locomotive and then transferred to a programmable logic controller on the waste package transporter. The programmable logic controller on the primary locomotive will monitor and check all of the transporter's operations and system performance parameters (CRWMS M&O 1997j, pp. 53–57).

The loading mechanism in the waste package transporter will be equipped with multiple positioning sensors and other types of sensors that provide feedback about the motor current usage, temperatures, radiation levels, and other parameters. A camera system will be mounted inside the transporter for remote viewing of loading and unloading operations.

The waste package transporter systems, particularly the loading mechanisms, must be very reliable. When empty, the transporter will be fully accessible for routine and preventive maintenance. The transporter's programmable logic controller-based control system will be mounted outside the protective shielding so that it can be accessed directly by maintenance personnel.

Emplacement Gantry Control System. The emplacement gantry is one of the most critical elements of the waste emplacement system. Its basic function and operation are discussed in Section 4.2.3.2

The emplacement gantry is designed to operate inside the high-temperature, high-radiation environment of the emplacement drift gantry operations inside the emplacement drifts will be controlled by operators at a remote control console. During gantry transporting operations in the main drifts and during periodic preventive maintenance service outside the emplacement drifts, operators will be provided with a portable control console for local control of the vehicle.

The emplacement gantry system must perform its intended functions in a safe and reliable manner. The design limits system complexity and incorporates verified and validated high-quality components and software; fault-tolerant dual-redundant programmable logic controller-based control systems; backup power and communication systems; and diverse hardware and software technologies. In addition, the redundant and backup systems will be physically separated.

As shown in Figure 4-47, Emplacement Vehicle On-Board Control System, the emplacement gantry's on-board control system is made up of multiple integrated subsystems. Several of these subsystems, such as the vehicle locomotion and braking systems, are critical to the successful operation of the gantry. As with the transport locomotives, the control system design for the emplacement gantry is based on a dual-redundant fault-tolerant programmable logic controller-based control configuration. The redundant programmable logic controllers also employ an input/output voting strategy. Software systems will be designed to enhance overall reliability by providing system

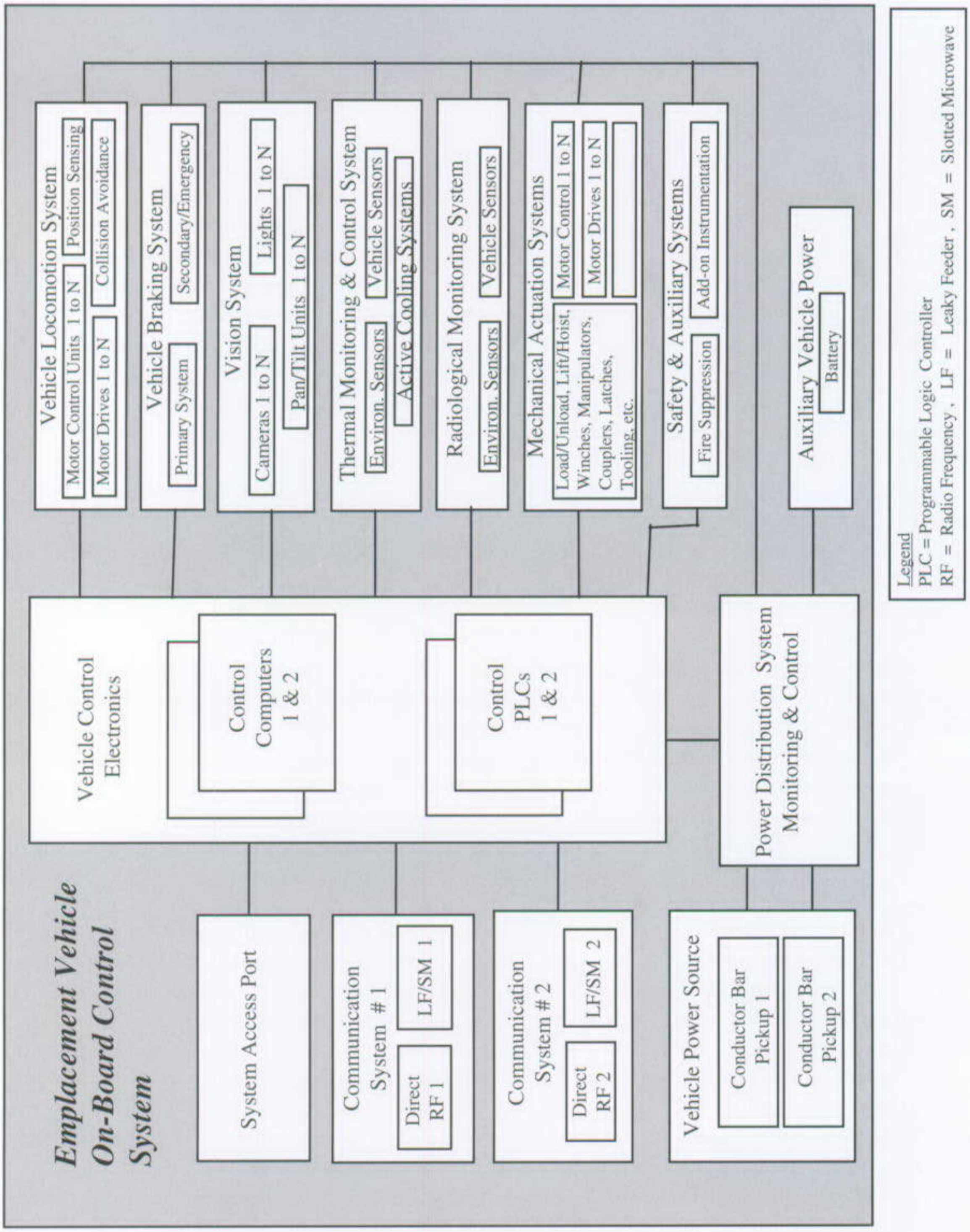

Figure 4-47. Emplacement Vehicle On-Board Control System

checks, monitoring system status, and reporting performance trends. Design strategies, such as using diverse technologies, physically separating redundant components, and providing backup power and communication systems, will be implemented to ensure safe, fault-tolerant operation.

The emplacement gantry's control system will be equipped with visual, thermal, and radiological sensors and instrumentation. These instruments will provide system operators at the remote control console with real-time feedback about the operating environment and vehicle performance. The vision system will consist of several on-board, radiation-tolerant articulated camera and lighting systems. The on-board thermal monitoring and control system will monitor the internal operating temperatures of the gantry and alert remote operators if temperatures begin to approach predefined operational limits. This system will also provide detailed graphical thermal profiles of conditions inside the emplacement drift. The on-board radiological monitoring system will continuously monitor the air for trace radionuclide gases that would indicate that a waste package had breached. The radiological monitoring system will also record cumulative dose radiation exposure of on-board electronics and other radiation-sensitive components. The on-board control system will alert operators of a malfunction or impending malfunction of sensors and instrumentation, allowing the gantry to be safely removed from the emplacement drift for troubleshooting and repair (CRWMS M&O 1997j, pp. 30–42).

Figures 4-48a and 4-48b depict a preliminary process and instrumentation diagram for the emplacement gantry. These diagrams identify electrical and software interfaces for all currently envisioned on-board instrumentation and control systems. Along with Figures 4-49 and 4-50, they show the level of detail developed for the VA emplacement gantry design. In the *Emplacement System Control and Communication Analysis* and the *Subsurface Waste Package Handling - Remote Control and Data Communications Analysis*, similar levels of design detail have been developed for the transport locomotives, waste package transporter, and the emplacement gantry carrier, as well as for the stationary control systems described in this section (CRWMS M&O 1997j and 1997ai).

Stationary Control Systems for Waste Emplacement. Stationary emplacement systems are the non-mobile waste transport and emplacement systems such as the emplacement drift isolation doors and the rail switches (Figure 4-46).

The emplacement drift isolation doors at the entrance of each emplacement drift must open and close each time access to the drift are required. The door assembly will be equipped with sensors and instrumentation that detect whether the door is open or closed and transmit this information to a remote monitoring station. As discussed in the *Emplacement System Control and Communication Analysis*, door position feedback can be provided using conventional mechanical limit switches, which can be configured to be redundant and fault-tolerant (CRWMS M&O 1997j, pp. 15–25).

A rotating beacon and auditory alarm system will be installed at the entrance of each emplacement drift. Before the doors open, the alarm horn will sound to warn personnel to vacate the immediate area. To ensure worker safety, the isolation door control system is also designed with local lock-out capability to prevent the remote operation of the door during maintenance or other activities near the doors. All of the operational inputs and outputs of an isolation door, including limit switches, actuators, auditory and visual alarm systems, will be connected to a programmable logic controller-based control node located in a junction box mounted on the drift wall near the isolation door. This approach will standardize and simplify component installation and wiring for each door. The door control system is designed to operate from a remote control console and to permit local control by personnel during maintenance operations. The doors will also be equipped with an emergency manual override capability.

As described in Section 4.2.3, the waste packages will be transported from the surface facilities to the emplacement drifts using a rail system. The rail switch control system network consists of a series of remotely monitored and controlled electrical

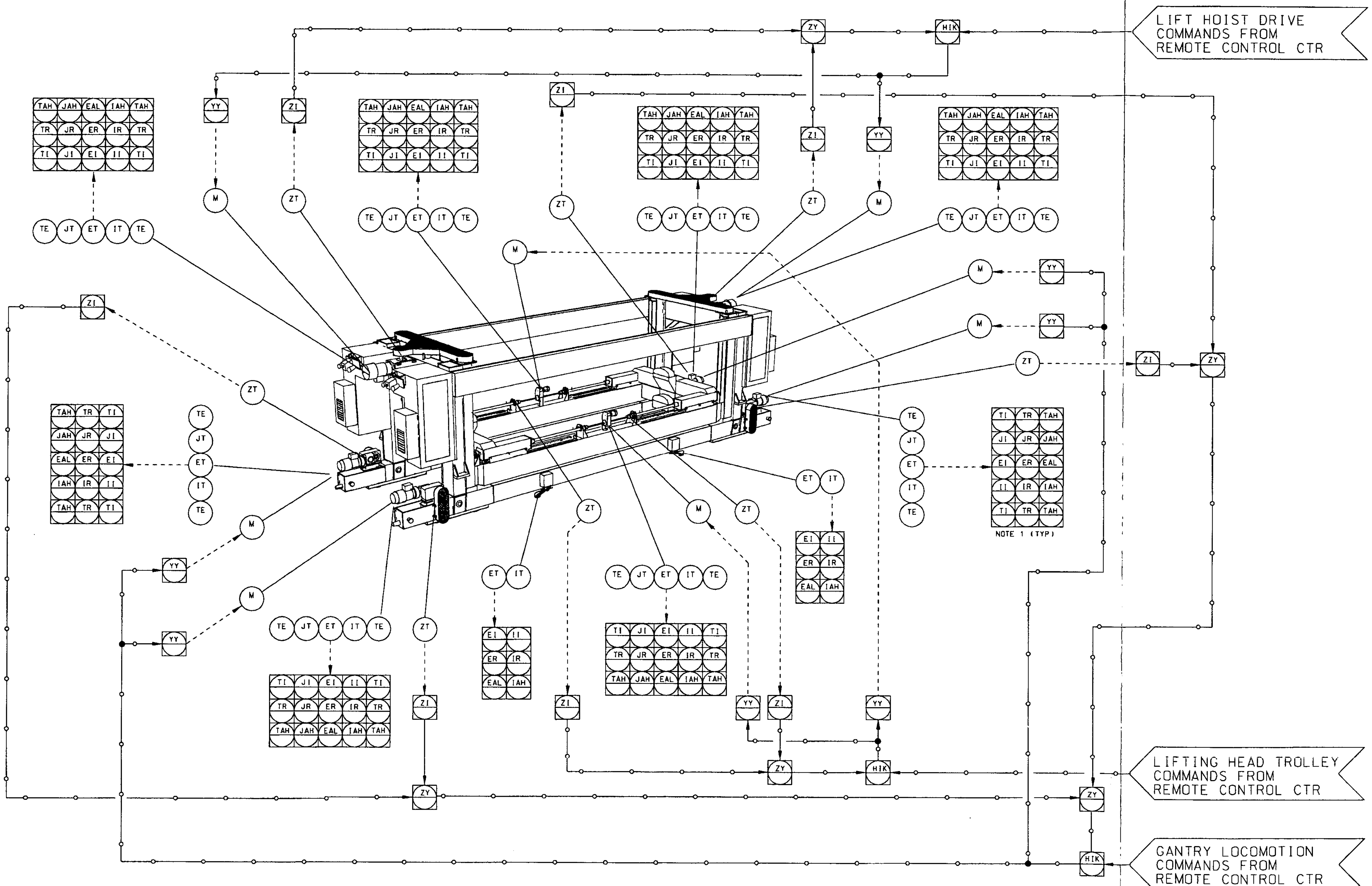

Figure 4-48a. Emplacement Gantry Control and Communication Piping and Instrumentation Diagram

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LEGEND:

EAL	VOLTAGE ALARM LOW
EI	VOLTAGE INDICATOR
ER	VOLTAGE RECORDER
ET	VOLTAGE TRANSMITTER
HIK	HAND INDICATING CONTROL STATION
IAH	CURRENT ALARM HIGH
II	CURRENT INDICATOR
IR	CURRENT RECORDER
IT	CURRENT TRANSMITTER
JAH	POWER ALARM HIGH
JI	POWER INDICATOR
JR	POWER RECORDER
JT	POWER TRANSMITTER
M	MOTOR
TAH	TEMPERATURE ALARM HIGH
TE	TEMPERATURE ELEMENT
TI	TEMPERATURE INDICATOR
TR	TEMPERATURE RECORDER
YY	EVENT CONVERTER
ZI	POSITION INDICATOR
ZT	POSITION TRANSMITTER
ZY	POSITION COMPUTER

*	INSTRUMENT ABBREVIATION
	SHARED DISPLAY MONITORING OR CONTROL POINT ON THE INSTRUMENT DATA HIGHWAY DISPLAYED AT THE SURFACE CONTROL CENTER AND LOCAL CONTROL STATION IF REQUIRED
	FIELD MOUNTED INSTRUMENT
-----	ELECTRIC SIGNAL
	SYSTEM LINK (SOFTWARE)
	SIGNAL INTERCONNECTION

NOTES:

1. INDICATION POINTS AND ASSOCIATED ALARM POINTS WILL BE INTERLOCKED WITH THE MOTOR CONTROL LOGIC TO PREVENT INADVERTENT OR UNSAFE OPERATION OF THE MOTOR.
2. THE LOGIC SHOWN IS FOR THE GANTRY LOCOMOTIVE, HOIST, AND TROLLEY DRIVE SYSTEMS. ADDITIONAL DRAWINGS WILL BE REQUIRED TO DEPICT THE FOLLOWING:
   - ENVIRONMENTAL MONITORING (TEMPERATURE & RADIATION)
   - INTERNAL EQUIPMENT MONITORING (TEMPERATURE & RADIATION)
   - GANTRY BRAKING SYSTEM
   - CAMERA CONTROLS FOR PAN/TILT/ZOOM
   - AUXILIARY POWER SYSTEM
   - FIRE SUPPRESSION SYSTEM
3. INSTRUMENT NUMBERING TO BE ADDRESSED AS DETAILED DESIGN IS DEVELOPED.

REFERENCES:

1. EMPLACEMENT SYSTEM CONTROL AND COMMUNICATION ANALYSIS, DI BCA000000-017170-0200-00016, REV 00.
2. PRELIMINARY WASTE PACKAGE TRANSPORT AND EMPLACEMENT EQUIPMENT, DI BCA000000-01717-0200-00012, REV 00.
3. EMPLACEMENT GANTRY PLAN AND ELEVATIONS BCAF00000-01717-2700-85007 REV 00

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Figure 4-48b. Emplacement Gantry Control and Communication Piping and Instrumentation Diagram (Continued)

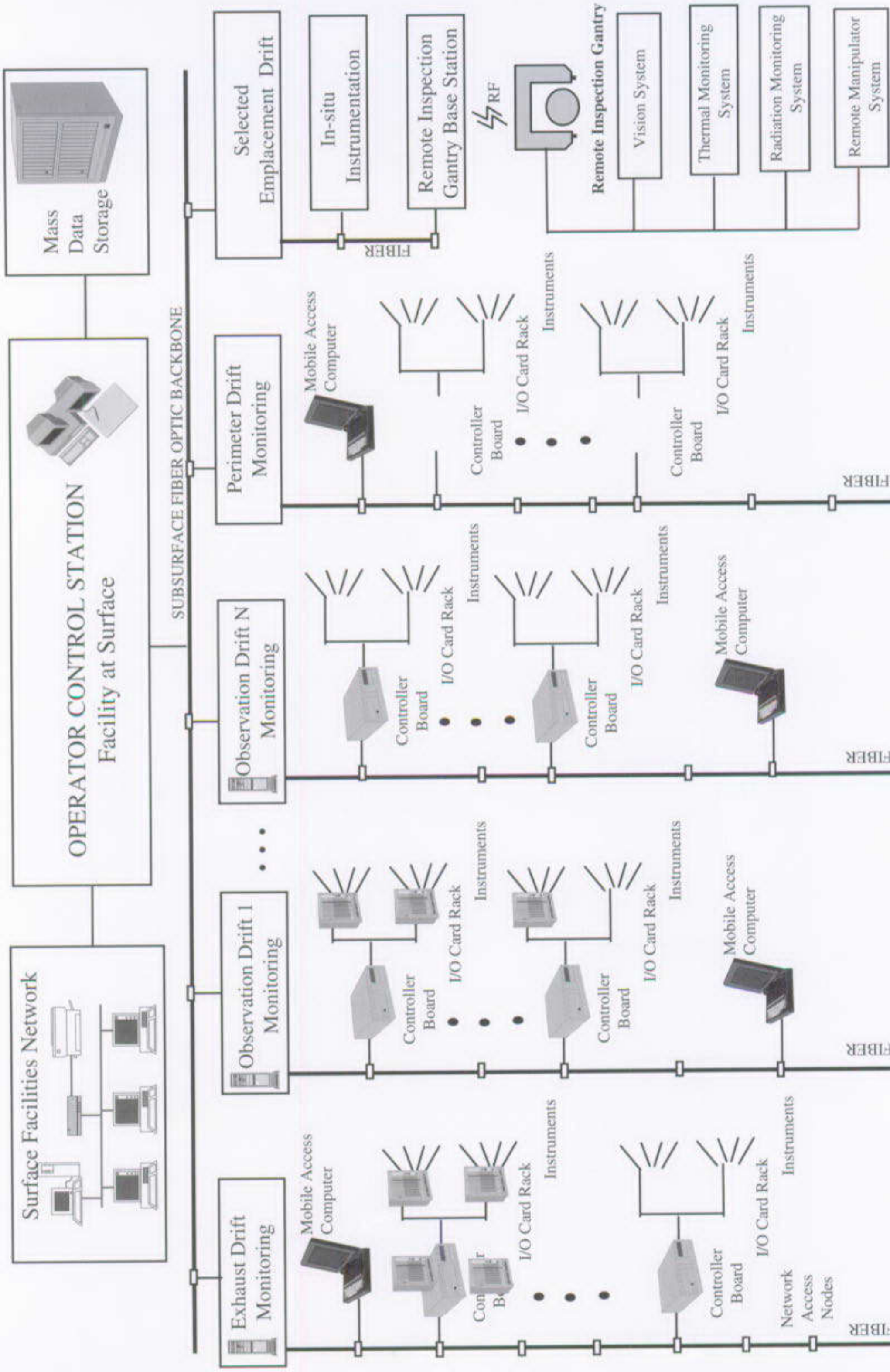


Figure 4-49. Monitoring and Instrumentation Diagram

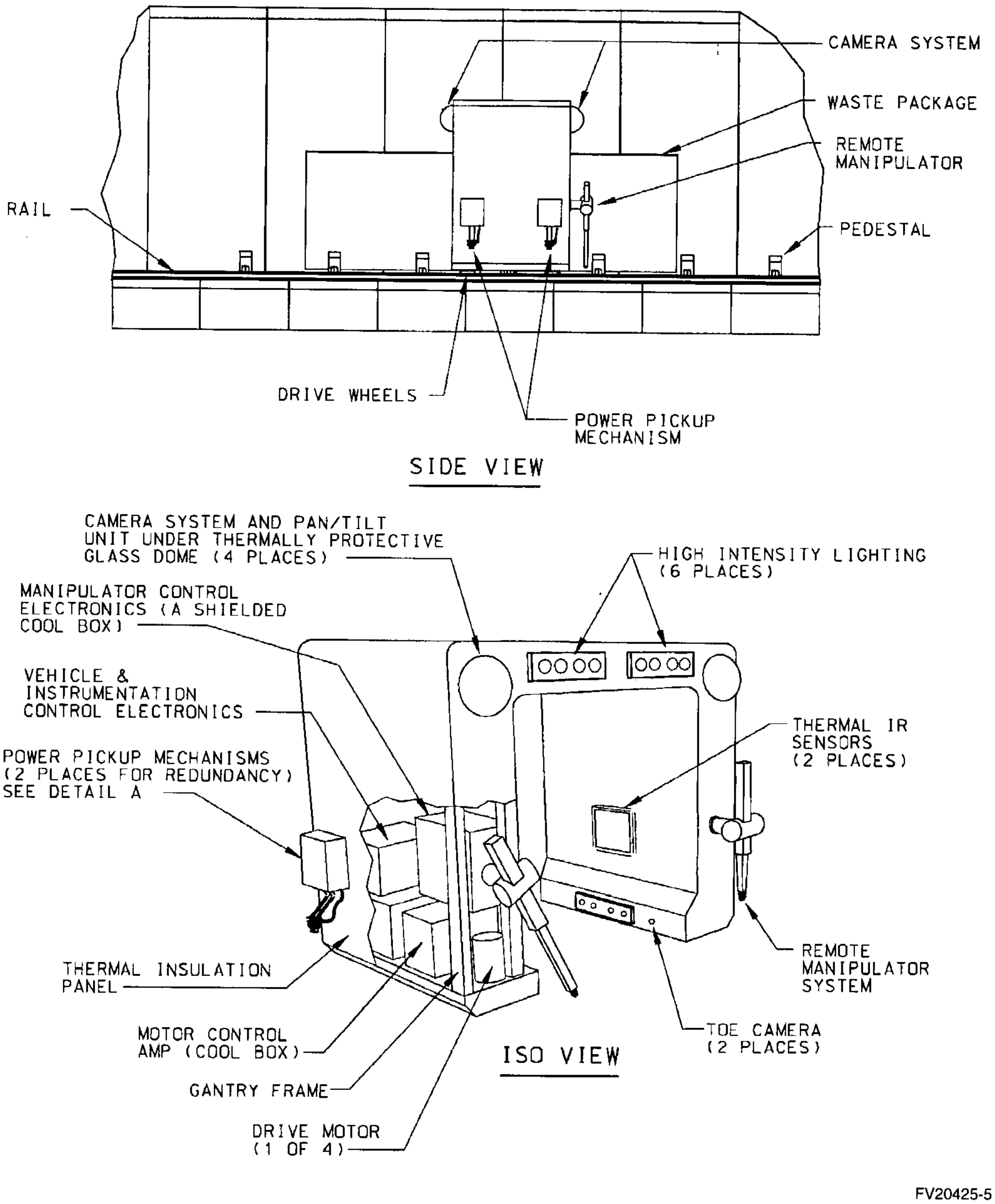


Figure 4-50. Remote Inspection Gantry

switching devices to activate the rail switches and divert the rolling stock. The capability for local control and actuation by personnel near the location of the switch, as well as a means for manual operation, are provided. A programmable logic controller-based circuit switch controller will electrically detect the position of the rail switch. The system will be equipped also with traffic lights and rail position indicators to alert the locomotive operators to the status of each rail switch (CRWMS M&O 1997j, pp. 26–28, CRWMS M&O 1997ai, pp. 15–18).

4.2.5.2 Emplacement System Data Communications Network

Subsurface repository instrumentation and control systems consist primarily of digitally based systems. These systems facilitate establishment of local area networks similar to those used in modern industrial facilities to provide plant-wide supervisory control and data acquisition capabilities (CRWMS M&O 1997j, pp. 54–61).

Data communication will provide for the monitoring and control and for the collection and dissemination of information throughout the subsurface repository. The data highways will also provide connection between operational control facilities located at the surface and digital monitoring and control systems in the subsurface.

Current design concepts call for the subsurface data communication highway to be based on a multiple-redundant fiber-optic backbone. The fiber-optic backbone will run throughout the subsurface main drifts. Devices such as monitoring instrumentation, control programmable logic controllers, computer workstations, and radio-frequency modems will be connected to the fiber-optic backbone.

The importance of the communication system has led to the development of architectures for data highway communication networks (Preckshot 1993). Several network configurations have been considered including star, bus, and ring configurations. The star configuration configures the network so that all devices interconnect through a series of central nodes, similar to a telephone switchboard. The bus configuration is configured in one long trunk onto which smaller branches are used to interconnect each station to the data highway. In the token-passing ring-based configuration, communicating devices are connected directly to a data highway forming a closed ring. Ring configurations are typically used in high-end computer networks and are inherently fault-tolerant. Current VA design concepts envision the use of a mixture of configurations; however, design work in this area is ongoing (see Section 3.2.1.5 of Volume 4).

The design and development of fault-tolerant, fail-safe local area networks and supervisory control and data acquisition networks has been accomplished in safety-related industries and are of importance to the successful design of the subsurface repository. Typical design strategies address issues such as the use of multiple-redundant data highways, real-time deterministic systems verses non-deterministic systems, and avoidance of common mode failures.

Initial estimates of the overall size and complexity of the monitoring and control system for the waste emplacement system indicate that there are approximately 6,000 data points associated with the monitoring and control of emplacement-related systems. Emplacement-related systems include the emplacement drift isolation doors, rail switches, and mobile equipment such as the emplacement gantry and transport locomotives (CRWMS M&O 1997i, pp. 62–63). This indicates that development of a reliable data communication system should be achievable using available commercial data communication technologies and practices.

In addition to the data communications systems, personnel working underground will use voice communications system to communicate within the subsurface facility and with surface facility personnel. The voice communication systems design, including a telephone system, a radio system, and a public announcement system, is currently under development.

4.2.5.3 Monitoring and Control Systems for Performance Confirmation

Performance confirmation is defined in 10 CFR 60.2 as "the program of tests, experiments, and analysis which is conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objectives for the period after permanent closure will be met." Subpart F of 10 CFR 60 provides general requirements for a performance confirmation program, as well as more specific requirements related to confirmation of geotechnical and design parameters, design testing, and monitoring and testing of waste packages.

The objectives of the performance confirmation program (CRWMS M&O 1997s) are to:

- Confirm that subsurface conditions encountered and any changes in those conditions during construction and waste emplacement operations are within the limits provided during the licensing process.
- Confirm that the natural and engineered systems and components that are required for repository operations, or are assumed to operate as barriers after permanent closure, function as intended and as anticipated (10 CFR 60.140). These systems and components specifically include the waste package, emplacement drift backfill (if used), and borehole seals.
- Evaluate compliance with regulatory and licensing requirements related to postclosure performance.
- Evaluate the repository's readiness for permanent closure.

Performance Confirmation Parameters to be Monitored. To achieve the objectives of the performance confirmation program, the performance confirmation data acquisition system will collect data and information partially represented by the following list:

- Air temperature and relative humidity in the emplacement drifts
- Type and amount of radioactive gases emanating from the emplacement drifts (the system will identify the specific drift that is the source of any radioactive gases)
- The condition of the waste packages and the emplacement drifts
- Placement and recovery of test coupons of sample materials in the emplacement drifts
- Groundwater flow into the emplacement drifts and evidence of standing water accumulating in the emplacement drifts
- Air permeability, in situ stress, and deformation and displacement of the rocks around the emplacement drifts
- Soil and rock temperature around the repository
- Moisture content, vapor content and humidity, fluid temperature, and air pressure in the altered zone

Performance Confirmation Data Acquisition Strategy. The performance confirmation data acquisition strategy will involve the following types and methods of collecting information:

- Geologic sampling and mapping
- Alcove-based testing in the non-emplacement area
- Borehole instruments in the altered zone
- Ventilation exhaust air monitoring
- Remote inspection systems deployment within emplacement drifts
- Possible recovery of waste packages for testing

Confirming the repository performance involves the integration of information from several diverse systems (Figure 4-49). This section addresses subsurface monitoring and control systems associated with the following:

- Ventilation exhaust air monitoring
- Remote in-drift inspection systems
- Repository monitoring from performance confirmation, non-emplacement observation drifts

Further information on other aspects of the performance confirmation data acquisition strategy is provided in the *Performance Confirmation Plan* (CRWMS M&O 1997s) and the *Performance Confirmation Data Acquisition Plan* (CRWMS M&O 1997r).

Exhaust Air Monitoring. The exhaust air monitoring instrumentation and data acquisition systems are designed to monitor radiation levels, temperature, and humidity within the ventilation system. These instruments and control systems will be located in the exhaust main and will provide periodic sampling of the exhaust air for each emplacement drift.

As indicated in Figure 4-49, the system hardware consists of a controller board, input/output card racks, input/output cards, and cables that connect the input/output card racks to the controller board. The temperature, humidity, and radiation sensors will be wired to the input/output cards that are connected to controller boards.

The controller board then will interface with higher level computer equipment that displays, trends, alarms, and stores the data. Controller boards are usually limited in the number of input/output racks they can support and the distances over which they can accurately collect data. Therefore, control nodes will be located at periodic intervals along the length of the exhaust main drift (CRWMS M&O 1997r, pp. 33–46).

Remote Inspection Systems. Preliminary VA design concepts call for suspending active cooling of individual emplacement drifts after each drift is filled with waste packages. Preliminary estimates indicate that the thermal environment inside the drifts may eventually rise above 160°C (320°F). The thermal and radiological conditions inside the emplacement drifts will prohibit direct access to the emplacement drifts by personnel for the duration of the monitoring phase.

A remotely controlled inspection vehicle will be deployed to monitor conditions inside the emplacement drifts for limited periods of time. This remote inspection vehicle, shown in Figure 4-50, will be capable of performing visual, thermal, and radiological inspections. It will also deposit and recover sample materials (test coupons) and recover dust samples (CRWMS M&O 1997r, pp. 47–82). The inspection vehicle will survey and monitor the conditions of the waste packages, drift lining, invert, support assembly, and other in-drift structures.

The on-board control system architecture is configured similar to that of the waste emplacement gantry (see Section 4.2.5.1 and Figure 4-47). Remote communication with the remote inspection vehicle will use the same systems used for communicating with the emplacement gantry: a distributed antenna system and either leaky-feeder or slotted microwave wave-guide technologies. Power for the remote inspection system will be supplied by the same third-rail, or conductor bar, system used by the waste emplacement gantry.

As shown in Figure 4-50, preliminary design for the remote inspection vehicle incorporates vision systems, thermal monitoring instruments, radiological monitoring instruments, air and gas sensing instruments, and remote manipulator systems. All of these vehicle systems are designed to operate reliably in the high-temperature high-radiation environment anticipated inside the emplacement drifts.

Initial thermal analyses indicate that it is technologically feasible to develop a remotely operated inspection vehicle that can operate in the high heat of the emplacement drift for up to 3 or 4 hours. After completing an assigned task, the remote inspection vehicle will be removed from the emplacement drift, allowed to cool, and prepared for the next task. All maintenance and servicing related to the remote inspection system will be performed either in human-rated areas in the subsurface or at a maintenance area on the surface.

Operators at a remote control console will control the inspection vehicle. These operators will be able to initiate operational commands via a graphical user interface and will be able to receive real-time visual and performance feedback from the vehicle. This will include position, speed, acceleration, braking status, internal and external temperatures and radiation levels, power usage, and status of auxiliary systems. Human operators will be in direct control of the inspection vehicle at all times and will be provided with direct, fail-safe, emergency shutdown capabilities. The control computers will continuously monitor every aspect of equipment operation and performance and will provide early warning signals and/or alarms to alert the operators if any parameter approaches predefined operational limits.

Performance Confirmation Drift (Observation Drift) Monitoring. The performance confirmation system includes monitoring and gathering data from borehole instrumentation within performance confirmation drifts or along the perimeter mains. Performance confirmation drifts will be constructed about 15 m (50 ft) above the emplacement drifts (as described in Section 4.2.1.2). Boreholes will be drilled from the performance confirmation drifts to approach the rock mass near the emplacement drifts. Selected instrumentation will be installed in the boreholes to gather data from the altered zone. Data acquisition hardware installed in the performance confirmation drift alcoves will transmit the data to the surface. As discussed in *Performance Confirmation Data Acquisition System*, instrumentation also will be installed in the test alcoves in the perimeter mains and in the cross-block drifts to gather data from the rock mass that lies between the waste packages and the saturated zone (CRWMS M&O 1997r, pp. 82–94).

Table 4-3 lists parameters to be monitored from the boreholes drilled in the performance confirmation drifts and test alcoves. These parameters do not represent a complete or final list; however, they are representative of the thermal, mechanical, hydrological, and chemical characteristics to be monitored for performance confirmation. For each parameter, an instrument type is given to illustrate a reasonable measurement approach. The range of the actual readings has not been established for specific instruments. However, if a typical instrument is known, it is listed. For these typical instruments, the stated operating range and output are given to provide information supporting data acquisition system planning. In many cases, where a typical instrument is cited, some modification or new development may be necessary to fulfill the objectives of performance confirmation monitoring. For example, system designers must consider longevity requirements and whether an instrument needs to be replaceable or should instead be designed as a probe rather than as a permanent installation.

Operating temperatures (rock and air temperatures) are expected to be the most important constraint on any instrumentation installed near the emplacement drifts. For example, temperatures of the rock mass may range from ambient to an allowable upper limit of 200°C (392°F). Performance confirmation drift air temperatures may range from ambient to an allowable limit of 50°C (122°F) (dry bulb).

This information and numerical values are preliminary, and are meant only to show examples of typical existing instrumentation and do not necessarily match the design needs for a given installation.

4.2.5.4 Additional Subsurface Monitoring and Control Systems

Ongoing design work is focused on development of the overall repository-wide integrated control system. Design strategies are being developed to address hardware and software issues related to using modern digital instrumentation and controls (see Section 3.2.1.10 of Volume 4).

Table 4-3. Typical Instruments for Observation Drift and Borehole Monitoring

Parameter	Location	Instrument Type	Example Instrument/ Manufacturer	Example Temperature Range
Strain/stress change at a point	Borehole	Vibrating wire stress meter	Geokon	-30 to 65°C
Rock mass displacement	Borehole	MPBX (multi-point borehole extensometer) Deflectometer / In-place Inclinometer	Geokon Geokon	-20 to 80°C -20 to 50°C
Air temperature (dry and wet bulb)	Performance confirmation drift	Humidity sensor	Vaisala HMP 235	-40 to 180°C
Rock temperature	Borehole	Thermocouple RTD Rapid Evaluation of K and Alpha-Thermal Probes (REKA)	Pyromation Type K Watlow Gordon 4-wire UNR	-270 to 1372°C -200 to 650°C 0 to 260°C
Rock moisture	Borehole	Humidity sensor Neutron logging ERT (electrical resistivity tomography)	Vaisala HMP 235 Century Geophysical Advanced Geosciences	-40 to 180°C up to 105°C up to 150°C
Relative humidity/dry bulb air temperature	Performance confirmation drift	Humidity sensor	Vaisala HMP 235	-40 to 180°C
Rock pore air pressure	Borehole	Air-K testing with packer system	SEAMIST system with pressure transducer	up to 160°C
Water chemistry/mineralogic changes	Borehole	Borehole chemical sensor and fluid sampling	SEAMIST system with chemical sensors	up to 160°C

Design work to evaluate the needs and functionality of a central monitoring and control facility has begun. This centralized command and control facility may be used for monitoring and controlling subsurface repository systems and status including the ventilation system, radiological monitoring systems, and the status of fire protection systems. This central control facility may also house the remote control consoles for waste emplacement and performance confirmation activities. In addition, it can provide a centralized location for monitoring the location of all personnel working in the subsurface and for accessing subsurface transportation systems, safety and security systems, and closed-circuit television systems. High-level monitoring and control of subsurface utility systems such as electrical power, water distribution, compressed air, and lighting systems may also be performed at a centralized control facility.

The central command and control facility may also provide integration and coordination for emergency response operations and access to subsurface repository communication systems including phones, wireless and mobile phones, two-way radios, public address and paging systems, and auditory alarm systems.

4.2.6 Utility Systems

The utilities needed for repository subsurface operations are similar to those used in the Exploratory Studies Facility. They include the following:

- Water System
- Electrical System
- Compressed Air System
- Communications System

The utilities for construction and development operations will be removed as these operations are completed. The utilities for emplacement operations will remain throughout the operating life of the repository. Repository openings will be sized to accommodate installation and maintenance of utilities without interfering with other activities. As discussed in the *Repository Subsurface Layout Configuration Analysis*, the utility requirements for each phase of operations will be determined in the future (CRWMS M&O 1997ab, Section 7.1.10). Future work will address the required utility systems as discussed in Section 3.2.1.9 of Volume 4.

Each side of the main drift isolation barriers will have utilities for specific functions. These utilities may include supply and wastewater pipelines, a compressed air pipeline, electrical cable(s), and monitoring and control cables.

Currently, there is no plan to extend electrical cables, pipelines, and ventilation ducts through the isolation barriers. Utilities may terminate at or near the isolation barriers on either side. If any utilities are passed through the barriers, transition connections will be provided in the barrier-covering panels or cast-in-concrete barriers, as discussed in *Subsurface Ventilation Isolation Barriers*, (CRWMS M&O 1997ah, Section 7.2.3).

Underground construction and development activities at the repository require both potable water (for drinking) and nonpotable water. Nonpotable water will be used for the following activities:

- Rock excavation equipment
- Washing walls for geologic mapping and before placing concrete segments and cast-in-place lining
- Dust suppression
- Fire suppression

The development shaft and the emplacement shaft construction activities will require nonpotable water for the following:

- Raise borer
- Shaft down reamer
- Placement of the concrete lining

Emplacement operations will also require potable water for drinking and nonpotable water for operations and fire suppression. The quantities of water required for these activities have not been determined.

It is anticipated that the existing Well J-13 and its associated pumps have adequate capacity for repository construction and operational water needs. Additional tanks, booster pumps, and pipelines may be required to accommodate increased construction or emplacement activities and storage needs. Well J-12 and C-Well can provide standby capacity to cover maintenance and emergency conditions at Well J-13.

4.2.7 Waste Retrieval System

Information contained in this section discusses aspects of the NRC Key Technical Issue on Repository Design and Thermal-Mechanical Effects (NRC 1997c). A description of the key technical issue is located in Section 4.3.3.6 of Volume 4. The stability of the underground excavations and the thermal effects were a consideration in the development reference design of the waste retrieval system. DOE is in the process of testing the materials of the ground control systems discussed in Section 4.2.2.2 under a variety of thermal conditions and will continue to consider these aspects as the design matures as described in Section 3.2.1.6 of Volume 4.

While 10 CFR 60 requires that the emplaced waste packages be retrievable for 50 years from the time that emplacement operations begin, DOE has taken a conservative approach and extended the duration to 100 years, as discussed in Section 3.1.4. Except where otherwise noted, the description of the waste retrieval system presented below is based on the *Retrievability Strategy Report* (CRWMS M&O 1997ad). This section describes how the waste packages will be retrieved under normal and off-normal conditions. Normal conditions are those conditions that are expected: the drift and rail system are intact, and communications and power systems are operational. Off-normal conditions are those in which these expected conditions are not met. Removing emplaced waste packages for other purposes, such as performance confirmation inspection, redistribution of inventory for ventilation purposes, relocation or transport to another area of the repository (surface or underground), is an operations function and is not considered retrieval, nor is it governed by any regulations or requirements for retrieval.

The VA reference design includes seven retrieval scenarios. These seven scenarios were developed in accordance with the requirements of the Nuclear Waste Policy Act, as amended, and 10 CFR 60. The following considerations were used to create each scenario:

- Reason for retrieval
- Retrieval conditions
- Whether full or partial retrieval is required
- Disposition of retrieved waste

4.2.7.1 Retrieval Scenarios

Table 4-4 summarizes the seven retrieval scenarios. Locating the repository host horizon above the water table in the unsaturated zone facilitates waste retrieval operations. This table shows three reasons for full retrieval (total system performance failure), as well as the four reasons for partial retrieval (poor waste package performance, unsuitable localized areas limiting emplacement, and resource recovery). Total system performance failure means that the system is still intact (emplacement drift ground support, drift infrastructure, and waste packages), but for some reason, imminent failure of the system is anticipated. Retrieval conditions (normal or off-normal) are also identified. Each of the seven scenarios is accompanied by remarks about the critical aspects of retrieval for that particular scenario based on the *Retrievability Strategy Report* (CRWMS M&O 1997ad, Section 7.7); and *Waste Package Retrieval Equipment* (CRWMS M&O 1997ar, Section 7.2).

Each scenario was evaluated to estimate the amount of time required to retrieve the waste packages. For the full retrieval scenarios, an estimated 7–12 years would be required to retrieve the waste. The costs associated with full retrieval are not included in the VA cost estimate in Volume 5.

4.2.7.2 Retrieval Strategy

10 CFR 60.111(b) requires that waste be retrieved from a drift in about the same amount of time as it took to emplace the waste. Final determination of the equipment and procedures to be used will continue to be evaluated as the design for the underground facility matures. This will allow the actual conditions to dictate the retrieval strategy. Retrieval strategy is also influenced by the method used to emplace the waste.

To establish an acceptable level of confidence that retrieval is possible, proof-of-principle demonstrations will be completed and documented before the license is issued to receive and possess waste. A test and evaluation analysis will identify those components for which supplier data cannot demonstrate performance adequately. These components will then be tested to provide reasonable assurance that the planned retrieval method will function under off-normal conditions.

For the VA design, the requirement for proof-of-principle data and tests will be addressed by analyses and logic to demonstrate that waste packages can be accessed and moved during repository operations. No retrieval demonstration is required. The draft *Civilian Radioactive Waste Management Program Plan* (DOE 1994) modified DOE's position on development of the Monitored Geologic Repository to be consistent with congressional budget direction and restated the requirement for proof-of-principle before LA. This document states, "the License Application design will describe designs in enough detail to demonstrate operating safety and enable compliance reviews by the Nuclear Regulatory Commission." This design adequacy can be demonstrated in several ways for the VA. Proof-of-principle testing can be deferred until after the LA has been submitted.

The Monitored Geologic Repository will accommodate retrieval or movement of waste packages emplaced at either end of an emplacement drift.

Table 4-4. Summary of Retrieval Scenarios

Scenario Number	Reason for Retrieval	Retrieval Conditions	Full or Partial Retrieval	Remarks
1	Public health & safety (total system performance failure)	Normal	Full retrieval	Normal retrieval methods will be used.
2	Resource recovery	Normal	Partial retrieval	Involves retrieval of only the waste packages containing valuable fissile material. Retrieving waste packages concurrently with emplacement increases demands on the ventilation system. Operations under this scenario will be scheduled to allow the drift to cool during the retrieval and emplacement process.
3	Public health & safety (poor waste package performance)	Off-normal (mechanical breach-radiation release)	Partial retrieval	All of the solutions to off-normal retrieval problems depend on whether at least one hole in the waste package skirt is accessible for attaching retrieval equipment, and whether the structural integrity of waste package will allow it to be pulled onto a slightly inclined plane. Damage to a waste package that makes the holes inaccessible, or that prevents the container from being dragged, is not anticipated.
4	Public health & safety (limited emplacement area unsuitable)	Off-normal (water inflow-limited area)	Partial retrieval (more than 120 waste packages)	Although the presence of water in the emplacement drift would be considered off-normal, any amount of water that might enter is expected to be minimal and would not preclude the use of normal retrieval methods.
5	Public health & safety (poor waste package performance)	Off-normal (exceeds expected emplacement drift temperatures)	Partial retrieval	Debris plugging the main drift or emplacement drift would be removed to restore ventilation. If that option fails, an alternate drift could be mined around the plug so that air flow could be restored.
6	Public health & safety (total system performance failure)	Off-normal (rockfall)	Full retrieval	If rockfall occurs in several areas, the scenario assumes that each rockfall can be handled as a localized event, and that there is no cumulative impact from several events that requires a separate scenario. The collapse of a very large area is considered improbable and is not included in this scenario.
7	Public health & safety (total system performance failure)	Off-normal (backfill)	Full retrieval	Retrieval is a preclosure activity. Backfilling, if done, would be a closure activity. As such, retrieval will not occur after backfilling.

Source: *Waste Package Retrieval Equipment*, Table 7.2.2.1 (CRWMS M&O 1997ar)

The VA design includes a rail system that extends the full length of the emplacement drift, from the east main to the west main drift. With this layout, emplacement equipment will be able to access the waste packages from either main drift. This capability enhances flexibility both in accessing waste packages for repository operations and in retrieving the waste packages under off-normal conditions, such as a rockfall. If a rockfall occurs or tracks are damaged, emplacement equipment (such as the emplacement gantry) will be able to access waste packages from the opposite main drift. After the damaged area has been repaired by specialized equipment, emplacement equipment can again use either main drift if the track is undamaged beyond the repair area.

The ventilation system will be capable of cooling one emplacement drift from a maximum of 200°C (392°F) to a maximum of 50°C (122°F) within 30 days. The maximum temperatures cited are for the temperature of the air as it enters the emplacement exhaust shaft. The ground control system, described in Section 4.2.2, is designed to support this rate of cooling.

A temporary storage facility capable of storing all of the emplaced waste planned for the repository will be sized and located on a plot plan before submitting the LA. Further evaluation of this site will be completed before issuance of the license to receive and possess waste. This site will be reserved throughout the preclosure period through repository closure. Design and construction of this facility will be deferred until it is needed to support retrieval.

4.2.7.3 Normal Retrieval Process and Equipment

The normal retrieval process, summarized in this section, is described in detail in the *Waste Package Retrieval Equipment* (CRWMS M&O 1997ar). Under normal conditions, the retrieval process is similar to the emplacement process with the steps involved for emplacement being carried out in reverse.

Emplacement Drift Preparation. Emplaced waste packages will cause the temperature within the emplacement drifts to rise. For retrieval, the drifts will be cooled. The main drifts will be ventilated throughout the preclosure period and air flow will be available throughout the retrieval period. The amount of time required to cool the drift will depend on how long the waste packages have been in place, the thermal load, the air flow rate applied to the drift, and the air intake temperature at the drift.

Air temperature will be measured until the air exiting the emplacement drift is at or below 50°C (122°F). After cooling, the remote controlled inspection vehicle, described in Section 4.2.5.3, will be used to inspect the emplacement drift to confirm that no debris obstructs retrieval equipment. Safety and health criteria require that a radiation survey be conducted to determine the level of airborne and direct radiation before retrieving the waste. This survey will identify potential hazards, provide data to estimate exposures, and permit the selection of proper protective equipment.

Any emplacement equipment that has been mothballed will be returned to working order.

Removing and Transporting Waste Packages. After the emplacement drift has been prepared, retrieval equipment will remove the waste packages from the emplacement drift and transport them to the surface. Under normal conditions, the equipment used to emplace the waste packages (Section 4.2.3) will also be used to retrieve the waste packages. The general steps required to remove the waste packages from the emplacement drift include the following:

- The gantry carrier will transport the emplacement gantry to the emplacement drift where the emplacement gantry will be unloaded.
- The emplacement gantry will move the waste package from the emplacement drift to the transfer dock area.
- The transporter doors and the emplacement drift isolation doors will be opened, the primary locomotive will position the waste package transporter at the transfer dock on the main drift side, the reusable railcar will be extended from the transporter, and the waste package will be loaded onto the reusable railcar.
- The loaded reusable railcar will be retracted into the transporter, the loaded waste package transporter will be moved away from the transfer dock, and the transporter and isolation doors will be closed.
- The primary and secondary locomotives will transport the loaded waste package transporter to the surface, where the waste

package is unloaded and the transporter checked for contamination.

- If additional waste packages are to be removed, the waste package removal steps will be repeated.

- The gantry carrier will remove the emplacement gantry crane from the emplacement drift.

4.2.7.4 Off-Normal Retrieval Process and Equipment

The retrieval process for off-normal conditions, summarized in this section, is described in detail in *Waste Package Retrieval Equipment* (CRWMS M&O 1997ar). Special equipment will be used when necessary, although existing operational emplacement equipment may be used for retrieval under off-normal conditions. For example, retrieval equipment for off-normal conditions could be used to clean up a rockfall, while emplacement equipment could be used to recover the waste package. The retrieval process for off-normal conditions is based on the following assumptions:

- When a waste package is damaged, at least one hole in the waste package skirt will be accessible (see Section 5.1.2.1).

- The waste package can be dragged onto retrieval equipment using one hole in the skirt.

- If a waste package is breached, the ventilation system will carry any contamination downstream, limiting the area of contamination.

- If the waste package transporter is not used, the retrieval equipment for off-normal conditions will transport the waste package to a predetermined facility on the surface to unload the waste package.

Off-Normal Retrieval Process and Conditions. There will be two levels of response to an event requiring off-normal retrieval. The first level will include initial measures required to protect personnel in the repository and the general public. Actions will be taken to control radioactive materials and permit prompt suspension of operations.

The second level of response will consist of the following six steps required to remove the waste packages affected:

- Conduct a radiological survey to determine the radiation, airborne contamination, and contamination on repository surfaces. The data collected will be used to develop a retrieval plan for the specific event, using the generic retrieval processes for off-normal conditions that were established before licensing.

- Establish access controls at the limits of contamination.

- Confine contamination. The confinement system installed in the subsurface repository will limit the spread of radioactive materials within unoccupied areas and prevent or limit the spread of contamination to the occupied areas. If a waste package has been breached, either the hole will be plugged or the waste package covered, if possible, to control further spread of contamination. If plugging or covering is not possible, the ventilation system will control radioactive effluents by providing continuous air flow from uncontaminated areas to potentially contaminated areas. Bulkheads with high-efficiency particulate air filters will be used to control and limit the spread of contamination.

- Collect additional data required to determine the non-radiological conditions and hazards, including a visual survey if possible.

- Develop a retrieval plan for the specific off-normal conditions encountered.

- Provide specialized equipment necessary and implement the off-normal retrieval plan. The plan will include ongoing measurements of radiological contamination and appropriate decontamination procedures.

These steps will not necessarily be performed in the order shown. For example, during the visual survey, it may be necessary to collect additional radiological data. The entire process of retrieval under off-normal conditions will be monitored by the subsurface performance confirmation monitoring equipment and the operations monitoring systems. Additional monitoring systems may be required, depending on conditions in the retrieval areas.

Retrieval under off-normal conditions may be created by events described in the *Preliminary MGDS Hazards Analysis* (CRWMS M&O 1996d) or conditions created by deterioration of repository systems during the preclosure period. These events will be classified based upon location of the waste package, the presence or absence of radiation shielding for personnel protection, and the presence or absence of a breach in the waste package. The location and condition of the waste package will dictate the type of equipment required for off-normal retrieval.

Classification of Events Caused by Deterioration. Systems that cannot be maintained will be susceptible to failure caused by deterioration. The systems left in the emplacement drifts with the waste packages may, over time, deteriorate beyond usage. These systems include the transportation rails, the power system for emplacement and retrieval equipment, the communications system, the ground control system, the waste package support assembly, and the inverts. Retrieval equipment to be used in the emplacement drifts during off-normal conditions will have the following assumed operating restraints:

- Because rail systems may be unusable, off-normal retrieval equipment will be primarily wheel- or crawler-based steerable vehicles. The emplacement drift concrete or steel inverts will provide an even surface on which the retrieval equipment can operate.

- Because alignment normally provided by the waste package supports may no longer be present, retrieval equipment must be capable of accessing the drift areas from side to side while being maneuverable enough to square up with the waste package, regardless of its configuration.

- Because the emplacement system power source may not be available, even if the power source is operational, a short extension tether, or umbilical, will be required to allow the third-rail power system to operate with the wheeled or tracked equipment.

- The retrieval equipment may be controlled from a system other than the emplacement drift data communications system, if that system is damaged.

The equipment and process employed must be practical for retrieving many waste packages over the time that it takes to correct any deficiency in the emplacement drifts, such as shielding the waste packages during transport to the surface.

Description of Off-Normal Retrieval Equipment. The types of equipment selected to perform off-normal retrieval will be based on the particular capabilities of the equipment. Some equipment will be standard, such as the heavy-duty forklift; some equipment will be custom-designed for specific functions, such as the inclined plane hauler. An extensive discussion on the types of events, condition and location of the waste package, and the equipment needed for the off-normal retrieval is contained in the *Waste Package Retrieval Equipment* (CRWMS M&O 1997ar, Section 7.2.4).

4.3 CLOSURE AND DECOMMISSIONING

The closure phase will begin after NRC amends the license to authorize closure of the repository. The closure phase is expected to span about 6 years, and may begin as early as the year 2110. The postclosure repository area will conform to the plans approved by NRC as part of the license amendment for permanent closure. Information contained in this section discusses aspects of the design and long-term contribution of repository seals in meeting the postclosure performance objectives as part of the NRC Key Technical Issue on the Repository Design and Thermal-Mechanical Effects (NRC 1997c). A description of the key technical issues is located in Section 4.3.3.6 of Volume 4. As noted in the following discussion, there has been some preliminary work performed on the design of the location and material section of the seals. As discussed in *Mined Geologic Disposal System Concept of Operations*, permanent closure of the repository will include closing the subsurface facilities, decontaminating and decommissioning the surface facilities, reclaiming the site, and establishing institutional barriers (CRWMS M&O 1997o, Section 3.4). The activities that will be involved are summarized in the following list:

- Closing the underground openings will include final backfilling of the boreholes and open operational areas within the underground facility that remain after waste emplacement has been completed. Closure operations will include removing underground equipment, backfilling underground openings, and sealing boreholes, shafts, and ramps.

- Decommissioning surface facilities will include permanently removing surface facilities and components that were necessary for preclosure operations. These facilities will be removed only after repository closure, in accordance with regulatory requirements and environmental policies. Decommissioning will include facility decontamination, dismantlement, and removal.

- Reclaiming the site will include actions taken to restore the site, to the extent possible, to its original preconstruction condition.

- Establishing institutional barriers will involve implementing active and passive institutional controls to restrict access to, and avoid disturbing, the controlled area. The institutional controls will limit or prevent intentional and unintentional activities in and around the closed repository.

A Review of the Available Technologies for Sealing a Potential Underground Nuclear Waste Repository at Yucca Mountain, Nevada (SNL 1994), was conducted to determine whether or not the shafts, drifts, and boreholes could be closed using existing technology and material. The scope of the study included reviewing selected backfill and sealing case histories. The study also included visiting sites where various technologies had been used to determine whether the technology needed to backfill and seal an underground repository existed and to identify any deficiencies associated with those technologies. The study concluded that existing technologies were adequate for the materials and placement methods being considered for backfilling and sealing the underground repository openings (SNL 1994, Section 6.0).

Backfilling Operations. Backfilling will be performed throughout the ramps, shafts, and main service drifts as part of the repository closure. These operations will require removal of equipment, rails, utilities, and unsuitable materials from the subsurface facility. Temporary utilities and support features, such as ventilation ducts, will be installed during backfilling and sealing and removed as backfilling progresses. The openings will be prepared to receive backfill by installing utilities and mobilizing equipment specifically dedicated to backfill operations.

Backfill and seals will be placed in a series of parallel operations that begin with placement of backfill in the perimeter mains adjacent to the waste emplacement drifts and continue through closure of shafts and ramps. As discussed in the *Mined Geologic Disposal System Advanced Conceptual Design Report*, backfilling will begin at the portions of the repository farthest from the openings and will continue to the surface openings. Sufficient access and ventilation to support workers and equipment will be maintained (CRWMS M&O 1996b, Volume II, Section 9.4.3). The ventilation system that was established for the monitor phase will be modified as the main drifts are plugged.

Surface Material Handling. Backfilling operations will require material handling support at the surface ranging from obtaining raw backfill material from the surface stockpile or other source, processing (screening, crushing, and possibly washing) the material to obtain the required particle size, and placing the processed backfill material into a stockpile for subsequent loading. No surface-related backfill designs have been developed because many design factors are unknown. Unknown factors include the extent of material degradation and settlement in the surface stockpile after 100 years of storage, whether single- or multiple-component fill material will be used, and the backfill emplacement rate. Material handling equipment at the surface will include loading, hauling, and processing equipment. Haul trucks will transport backfill to the shaft openings.

Using rock excavated during repository construction for backfill is being considered. However, after 100 years of surface exposure, the excavated rock may need to be sterilized by washing and chemical treatment before being used as backfill (CRWMS M&O 1996b, Volume II, Section 9.4.3.1.1).

Underground Material Handling. Backfill material will be transported underground by open gondola rail-cars. Backfilling will likely be performed concurrently at multiple locations to reduce the time required to close the repository. Approximately 2.6 million cubic meters (3.4 million cubic yards) of material will be required to backfill the main drifts, shafts, and miscellaneous underground excavations. Backfilling operations can be maintained at up to two or three locations at a time. To supply multiple placement locations, two transfer points will be installed to unload gondola cars and to load work cars (CRWMS M&O 1996b, Volume II, Section 9.4.3.1.2).

Backfill Placement System. A pneumatic system is the preferred method of placing backfill into the main drifts and ramps, and around ramp seals. A pneumatic backfilling system typically includes an air compressor or blower, stower, hydraulic drive unit, electrical power feeder and switchgear, material receiving hopper, and pipeline. The stower will be the central component of a typical industrial pneumatic backfilling system. The stower is a large rotary air lock that introduces coarse abrasive materials into a fast moving, low-pressure air stream. Eight compartments will be formed by elongated plates mounted to a central shaft that turns within a curved, tight-fitting case. The bottom of the stower will be vertically elongated, forming an enclosed trough through which compressed air flows. Fill material will be dumped into the top compartment, which is open to air, and will be carried into the base compartment. The rotor will be tightly fitted to the casement to prevent compressed air from escaping. Once suspended, the material will be conveyed through a pipeline connected to the bottom of the stower and sprayed into the targeted void.

The results of an informal survey of various field operations and documents describing backfilling applications performed in the 1970s and 1980s show the following:

- Blower sizes ranged from 110 to 630 kW at sea level, with 300 kW being most common, and produced air flows ranging from 1.4 to 2.8 m^3/s (2,965.2 to 5,930.4 ft^3/minute), with 1.9 m^3/s (2,120 ft^3/minute) being most common. Air pressures at the blowers ranged from 55 to 100 kPa with a pipeline operating pressure of 28 to 34 kPa. The blower speeds varied between 1,600 and 2,300 revolutions/minute.

- Pipelines included 8- and 10-in. diameter Schedule 40 and 80 steel pipe, rubber-lined light-duty construction pipe, and fiberglass construction pipe.

- Piping arrangements included vertical drops of up to 610 m (2,000 ft) and horizontal runs to 610 m (2,000 ft). Plugging problems became excessive in some cases when horizontal pipeline lengths exceeded 370 m (1,200 ft).

The VA design calls for the stower and material feed equipment to be mounted on rail cars entrained with a material supply car or supply cars and locomotive. While backfilling, the stower car and one supply car may be positioned at the site of backfilling, while another supply car is shuttled by the locomotive back and forth to a material feed storage pile. This arrangement will provide flexibility for the system to move throughout the subsurface and place backfill at widely scattered locations. If the rail or other man-made materials are removed before backfilling, the stowing equipment will be mounted either on a crawler or a steel-tired unit.

Most of the pneumatic backfilling systems surveyed on an industry-wide basis used long pipelines to distribute the material. The mobile system, however, will eliminate the need for a long pipeline, which is expensive and subject to plugging, by using a relatively short pipe to aim the backfill at the point of placement. The pipe snout will be swiveled and elevated to completely sweep the drift opening cross-section.

As discussed in the *Mined Geologic Disposal System Advanced Conceptual Design Report*, other advantages of this system include the ability to move a large volume of material, the ability to place the material in less time, and the operational flexibility to handle unexpected conditions (CRWMS M&O 1996b, Volume II, Section 9.4.3.2.1)

Sealing Shafts, Ramps, and Boreholes. As discussed in the *Repository Seals Requirements Study*, the qualitative design criteria for seals for shafts, ramps and boreholes is provided in 10 CFR 60.134 (CRWMS M&O 1997aa, Section 2.1.1):

- Seals for shafts and boreholes shall be designed so that, following permanent closure, they do not become pathways that compromise the geologic repository's ability to meet the performance objectives for the period following the permanent closure.

- Materials and placement methods for seals shall be selected to reduce the potential for creating a preferential pathway for groundwater and radioactive waste migration, through existing pathways, to the extent practicable.

The seals placed in the ramps, shafts, and boreholes will be strategically located to reduce radionuclide migration over extended time periods, and so that they do not become pathways that compromise the repository's postclosure performance. Seal materials and placement methods will be selected to reduce, to the extent practicable, the creation of preferential pathways for groundwater to contact the waste packages and the migration of radionuclides through existing pathways. The seals likely will be integrated with backfill and will be bracketed by the backfill. Installing seals will involve preparing the underground openings to receive the seals, obtaining and transporting seal material, and constructing the seals.

A number of seal geometries may be applicable to the repository. These types of seals include inundation plugs, hydraulic fill containment bulkheads, abandonment bulkheads, consolidation bulkheads, and conversion bulkheads. These seals are defined as follows:

- Inundation plugs are installed in shafts, ramps, and drifts to protect against sudden inflows of water. Inundation plugs often must withstand very large pressures and are usually designed for full hydrostatic head produced by elevated water table at the site of the operation.

- Hydraulic fill containment bulkheads are used to retain backfill that has been stowed as a slurry during drainage or decanting of the water until moist backfill has consolidated.

- Abandonment bulkheads are installed to seal abandoned underground excavations and limit pumping or ventilation requirements. These bulkheads are designed to withstand hydrostatic pressure in wet conditions and to be explosion-proof in gassy conditions.

- Consolidation bulkheads are used to provide a protection barrier behind which grout curtains can be installed. These bulkheads may be constructed in a shaft to provide a stable platform upon which inclined, vertical grout holes may be drilled or constructed at a drift heading to maintain the structural integrity of the rock face during grouting through inclined, horizontal drillholes. Consolidation bulkheads are designed to withstand specific hydrostatic limits.

- Conversion bulkheads are used for underground excavations that have been adapted to gas storage. This seal is constructed of multiple components that are incorporated into a complex geometry to limit leakage.

Some of these seals may not be adequate for repository application because they are gas or liquid permeable. Variations of the inundation plug and conversion bulkhead may be the most applicable for sealing the repository if permeability becomes an issue.

Basic seal shapes for any application are parallel-sided, arched, or tapered. Parallel seals may be keyed into the surrounding rock, while arched and tapered seals are inherently keyed. The preparation of a keyway will require an over-excavated section at a typical shaft or ramp seal location. Installing a seal with a keyway will require more preparation and materials than unkeyed seals. While leakage may occur along the interface between seal and host rock, or through the host rock only, the seal geometries are used when the potential leakage through these pathways is insignificant.

The *Repository Seals Requirements Study* (CRWMS M&O 1997aa) was conducted to identify design requirements for the repository sealing system. The repository sealing study considered repository sealing to be defined as those components that would reduce potential inflows of water or air into the repository. An assessment of the need for sealing shaft and ramp components was performed by comparing the allowable flow goals from 10 CFR 60 to the anticipated flows from the shafts, ramps, and underground facility. The repository sealing strategy considered where, when, and how to seal the repository. The results of this study were transferred to the *Controlled Design Assumptions Document* (CRWMS M&O 1998b) and will be used in future design activities as discussed in Section 3.2.1.8 of Volume 4.

The report and other previous studies have identified tests that should be conducted for the backfill and sealing activity. These tests have been incorporated into the *Viability Assessment Mined Geologic Disposal System Test and Evaluation Plan*, (CRWMS M&O 1998m).

Surface Decommissioning. The surface facilities will be decommissioned and removed from service during repository closure. Decommissioning will include decontamination, removal/salvage of valuable equipment and materials, and dismantlement.

Facility dismantlement will include the dismemberment, distribution, or removal of facility systems, in whole or in part, from the site. The surface facilities will require demolition of reinforced structures after the fixtures and equipment have been removed. The potential for salvaging, recycling, and reusing equipment, materials, and fixtures will be considered. Surface facilities must also be removed to perform final site restoration. As discussed in *Mined Geologic Disposal System Concept of Operations*, facility removal will include the preparation and transportation of systems, structures, and components to offsite locations. (CRWMS M&O 1997o, Section 3.4.2)

Site Reclamation. Site reclamation will include restoring the site to as near its pre-construction condition as practicable. Reclamation may require the recontouring of all disturbed surface areas, surface backfill, soil buildup and reconditioning, site revegetation, site water course configuration, and erosion control implementation (CRWMS M&O 1997o, Section 3.4.3).

Institutional Barriers. Institutional barriers will include land records and warning systems to be placed around the repository to prevent human disturbance. Provisions may be added for postclosure monitoring. The controlled area and repository operations area will be identified by monuments that are designed, fabricated, and placed to be as permanent as practicable (CRWMS M&O 1997o, Section 3.4.4).

4.4 PERFORMANCE CONFIRMATION PROGRAM

The performance confirmation program includes tasks required to monitor repository system performance. The performance confirmation program includes elements of site testing, repository testing, repository subsurface support facilities construction, and waste package testing that extends from the pre-emplacement construction phase through the monitoring phase of the repository (CRWMS M&O 1997s).

Site testing includes subsurface geologic mapping and sampling, surface-based hydrologic instrumentation and monitoring, underground fault zone hydrologic testing, sample testing, thermal testing, instrumentation and monitoring, and general surface-based testing.

Repository testing includes in situ seal testing for ramp, shaft, and borehole applications as well as in situ design testing, near-field testing, and laboratory testing for hydrocarbons remaining in the repository.

Waste package testing includes offsite laboratory testing of waste package materials, in situ waste package monitoring, and testing of recovered waste package material specimens. The key parameters affecting waste package performance to be measured in laboratory testing include those associated with oxidation and aqueous corrosion. All corrosion degradation modes identified as important to either the outer or inner barrier of the waste package would be measured in long-term laboratory corrosion tests. Oxidation and corrosion products would be characterized. Waste package in situ monitoring would include monitoring a variety of parameters in the subsurface excavations and surrounding rocks, including gaseous radionuclides whose presence would indicate an early waste package failure, container surface temperature, and humidity.

Waste package material testing may involve the recovery of actual waste packages for inspection and testing purposes. Because such recovery and testing is extremely costly, it would be employed only as a contingency; that is, should a breached or damaged waste package be detected, it would be recovered and brought back to the surface facility for testing and rework.

Repository subsurface support facilities include the construction of subsurface test facilities and support concepts, which would consist of an integrated network of systems to monitor directly the emplacement drift environment; monitor the geologic, hydrologic, and geochemical conditions adjacent to the emplacement drifts; and monitor conditions surrounding the repository block. Subsurface support facilities would include observation drifts and alcoves near the emplacement area, remotely operated systems for emplacement drift monitoring, monitoring equipment in the exhaust ventilation system, and recovery of waste packages, as a contingency. A remotely operated visual inspection system would obtain visual records of waste package surfaces, drift inverts and walls, ground support systems, and drift collapse/rockfalls in the drifts following waste emplacement.

A remotely operated thermal inspection system will measure waste package wall temperature, temperature on the emplacement drift wall, and drift air temperature following waste emplacement. Remote radiological inspection will monitor radiation levels in the emplacement drifts following waste emplacement to detect potential waste package failure and radionuclide release. Ventilation drift monitoring will provide observations of the incoming and outgoing conditions in the emplacement drift air, including temperature, relative humidity, and presence of gaseous radionuclides.

Data collected during the Performance Confirmation Program will be used to update the models used to evaluate total system performance. Results of monitoring and analysis could confirm predicted system response. If it were determined that actual conditions differed from those predicted, the results could support further evaluation of the impact of actual conditions on long-term performance of the repository system.

Systems and facilities required to conduct the performance confirmation program have been integrated into the VA reference design. Design of the performance confirmation drifts is described in Section 4.2.1.2, and monitoring and control systems are described in Section 4.2.5.3.

5. DESIGN OF THE ENGINEERED BARRIER SYSTEM

This section describes the VA reference design for the engineered barrier system. The engineered barrier system has the following two major components: the waste packages that hold the waste and the underground facility (as defined by 10 CFR 60) comprising those engineered features outside the waste packages. Additional features are being evaluated as design options that may be added to the engineered barrier system to improve performance. These features are discussed in Section 5.3. Figure 5-1 shows the engineered barrier system.

This section is divided as follows:

- Components and Design of the Waste Package
- Underground Facility
- Design Options for the Engineered Barrier System

5.1 COMPONENTS AND DESIGN OF THE WASTE PACKAGE

The disposal containers, or waste packages (loaded, sealed, and inspected disposal containers), are being designed to hold radioactive material that arrives at the repository in different forms and under different conditions. The following two paragraphs briefly state how the design of the waste package fits with other work being done to predict the repository's performance and describe which key technical issues the work supports. The remainder of this section discusses the waste forms to be handled, the designs of the various waste packages, and the extensive evaluations that have been made to predict the performance of the waste packages over time.

The waste package design contributes to reducing the uncertainties of the principal factors that impact preclosure operations and postclosure performance identified in Section 2 of Volume 4. As discussed in Section 2.2, the principal factors affecting the postclosure performance of the waste package include the following: water dripping onto the waste package, humidity and temperature at the waste package, chemistry of water on the waste package, integrity of the carbon steel outer barrier of the waste package, integrity of the high-nickel alloy inner barrier of the waste package, integrity of spent nuclear fuel cladding, and dissolution of the uranium oxide and glass waste forms. The waste package design addresses how these principal factors affect the lifetime of the waste package and its ability to isolate and contain the waste. This assessment is made using data generated in the materials testing programs that aid in selecting materials that perform well under the anticipated repository conditions. Once the materials are selected, design analyses are performed to evaluate the waste package's structural integrity, thermal performance, and shielding properties. In addition, the testing programs also provide data used to predict the containment time provided by the waste package barriers and the spent nuclear fuel cladding.

The waste package design and the materials and waste form testing also address various aspects of six of NRC's key technical issues as discussed in the following list:

- Structural Deformation and Seismicity (NRC 1997a). The waste package designs must be able to withstand structural deformation due to seismicity and related hazards to ensure that any damage would not pose a risk of noncompliance. Refer to Volume 4 Section 4.3.3.2.

- Evolution of the Near-Field Environment (NRC 1997b). The waste package lifetime is influenced by the amount of water contacting the waste package and the chemistry of the water. The thermal load must be considered for its effect on the waste and the natural environment. Refer to Volume 4, Section 4.3.3.3 for a description of the Key Technical Issue.

- Container Life and Source Term (NRC 1998a). A long-lived waste package is the goal of the waste package design. Structural analyses evaluate the integrity of

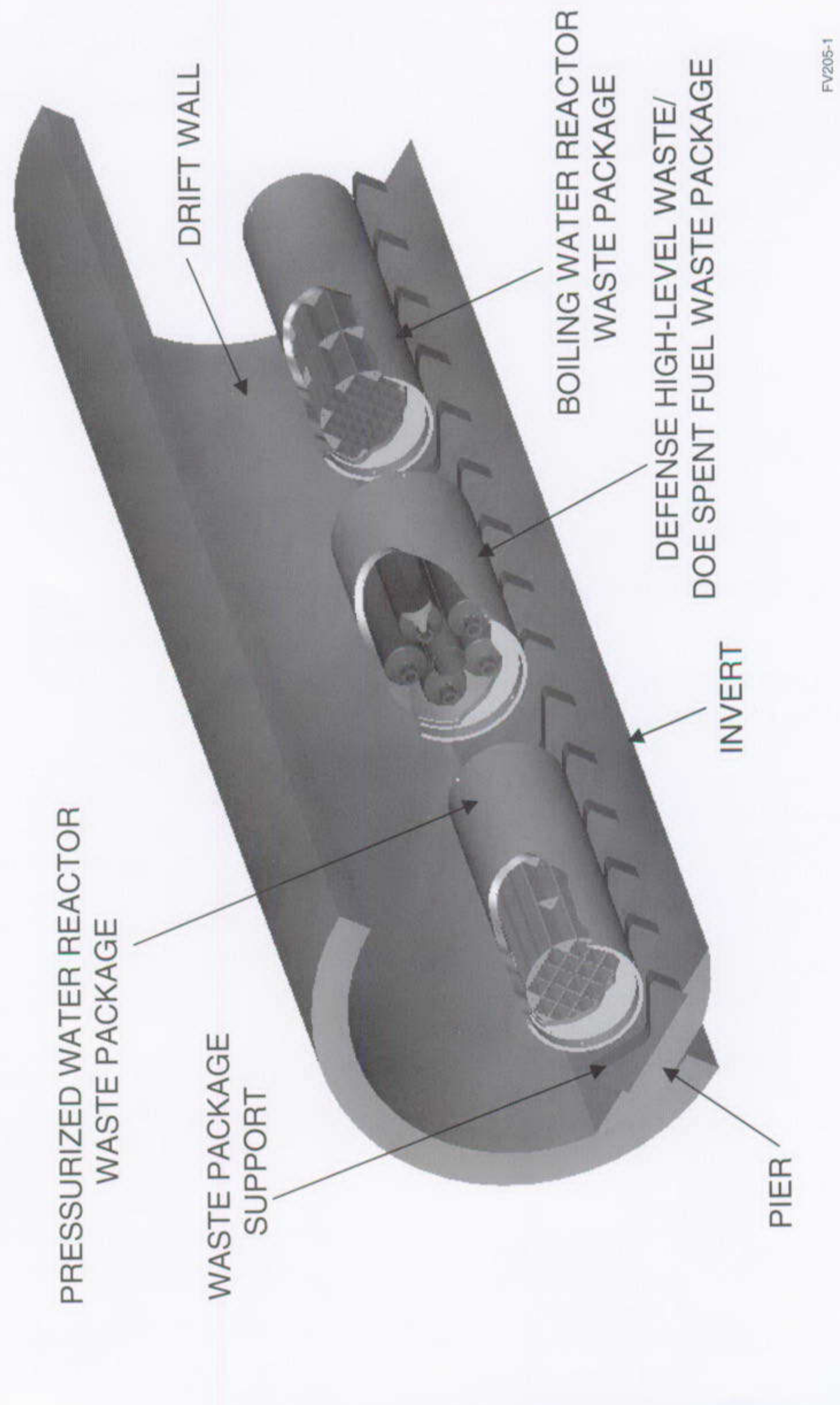


Figure 5-1. Engineered Barrier System

the waste package. Testing the waste package materials supports the generation of material corrosion or degradation models to evaluate the effects of corrosion on the lifetime of the waste packages. In addition, the testing of the waste forms provides source terms for radionuclide releases to the near-field environment. Refer to Volume 4, Section 4.3.3.4 for a description of the key technical issue.

- Repository Design and Thermal-Mechanical Effects (NRC 1997c). One of the subissues associated with this key technical issue is demonstrating that the designs for the repository and the waste packages meet the preclosure and postclosure performance objectives. Refer to Volume 4, Section 4.3.3.6 for a description of the key technical issue.

- Total System Performance Assessment (TSPA) and Integration (NRC 1998b). A TSPA will be conducted to determine compliance with limits set for dose and risk. The tests being done on the waste package materials and waste forms help determine their degradation modes under repository conditions. This information, in turn, is used as input to the TSPA to project offsite doses. Refer to Volume 4, Section 4.3.3.7 for a description of the key technical issue.

- Radionuclide Transport (Sagar 1997). This key technical issue is concerned with the rate at which radionuclides are released from the waste form and the extent of their movement away from the repository toward the accessible environment. The waste package materials and waste form tests contribute to understanding these processes. Refer to Volume 4, Section 4.3.3.10 for a description of the key technical issue.

5.1.1 Waste Forms

The waste forms can be grouped into the following four general categories: spent nuclear fuel from commercial nuclear reactors; wastes from nuclear materials processing (high-level radioactive waste); spent nuclear fuel from DOE programs, including naval spent nuclear fuel from the Naval Nuclear Propulsion Program; and surplus plutonium.

Spent fuel from commercial nuclear reactors refers to fuel coming from reactors that are used to produce power (electricity). There are two types of commercial power reactors in the United States-boiling-water reactors and pressurized-water reactors. The fuel for these reactors is arranged as assemblies, which are 6 to 17 ft in overall length. These assemblies are a collection of fuel rods arranged in square bundles and customized to meet the size and performance requirements of the reactor design. The fuel rods are seamless end-welded metal tubes, approximately 0.25 to 0.50 inches in diameter, which contain uranium dioxide, compressed into short cylinders called pellets. Most of the tubes are made of a special metal alloy, Zircaloy (about 99 percent); others are made of stainless steel (about 1 percent). A generic name for the material comprising the tubes is cladding (DOE 1996c, Appendix B; CRWMS M&O 1997as).

The cladding is designed and fabricated to perform in the harsh reactor environment where operating pressures can exceed 2,000 lbs/in.^{2} and fuel cladding temperatures approach 700°F (371°C). Both the Zircaloy and stainless steel cladding are high-strength metals exhibiting high tensile strength with wall thicknesses of about 0.7 mm. For purposes of conducting reactor safety analyses, it is assumed conservatively that 0.1 percent of the cladding is breached. This conservative assumption is carried over in the analyses performed for the performance assessment and is discussed in Volume 3. This assumed cladding failure rate of 0.1 percent is about twice that actually observed. Moreover, no credit for stainless steel cladding is taken in conducting performance assessments.

Approximately 300,000 fuel assemblies are expected to be generated by the year 2041—170,000 from boiling-water reactors (including about 300 with stainless-steel cladding) and 130,000 from pressurized-water reactors (including 2,000 with stainless-steel cladding). Current statutory limits on the total mass of heavy-metal, implemented through a license issued by NRC, allow about 75 percent or about 225,000 of these fuel assemblies to be emplaced in the first repository until a second repository is in operation (CRWMS M&O 1997av).

The high-level radioactive waste consists of solidified wastes from nuclear materials processing. This waste has various origins, such as residues from producing nuclear weapons and reprocessing fuel from commercial, research, and naval reactors, with the majority coming from noncivilian activities. The waste is solidified through a process that yields a leach-resistant material, typically a glass form called vitrified borosilicate glass. While still liquid, the glass is poured into stainless-steel canisters with an outside diameter of 0.61 m (24 in.) and a length of 3 or 4.6 m (9.8 or 15 ft). After the glass cools and solidifies, the canisters are sealed.

Approximately 19,000 canisters of high-level radioactive waste are planned to be generated by the year 2035. Approximately 1.5 percent of these canisters will be from reprocessing commercial nuclear fuel; the remainder will be from reprocessing materials from the defense nuclear program. Though the number of high-level radioactive waste canisters to be emplaced in the first repository has not been finalized, it is expected to be approximately 8,300 (CRWMS M&O 1997av).

DOE has more than 250 different types of spent nuclear fuel in its inventory (DOE 1997b). This fuel has various physical, chemical, and nuclear characteristics and represents approximately 2,500 MTHM (2,756 tons), of which 2,333 MTHM (2,572 tons), will be allocated for disposal in the first repository (CRWMS M&O 1997av).

The largest single component of this inventory is spent nuclear fuel from the N-Reactor with approximately 1,980 metric tons (2,181 tons) of heavy metal in inventory. During its 20-year life, the N-Reactor produced nuclear isotopes for defense purposes. This N-Reactor fuel is a uranium metal with a very low (less than 2 percent) enrichment. Enrichment is a measure of the initial concentration of uranium-235 in the fuel. The fuel will be placed in canisters that can be disposed of in the proposed repository. Approximately 400 canisters will be loaded with N-Reactor spent nuclear fuel (CRWMS M&O 1997av).

Approximately 180 metric tons (198 tons) of DOE heavy-metal inventory is low-enriched uranium oxide. Some of this material is standard commercial spent nuclear fuel that was used for testing. This fuel will be transported to the repository in standard transportation casks. Another fraction of this fuel is the fuel debris from the damaged reactor core at Three-Mile Island-2. The fuel debris is already in canisters that may be placed inside a larger container and transported to the repository (CRWMS M&O 1997av).

Approximately 125 MTHM (138 tons) are composed of highly enriched uranium (initially more than 20 percent uranium-235), medium-enriched uranium (initially between 5 and 20 percent uranium-235), and some thorium- and plutonium-based fuels. This fuel presents different considerations for avoiding criticality than does the commercial spent nuclear fuel, and therefore requires separate analysis. In general, these fuels will be placed in canisters. These canisters will be placed into waste packages along with canisters containing vitrified high-level radioactive waste; this process is called co-disposal. Some fuels within this group may require an additional waste package design either because the spent nuclear fuel canister is too large, or because there are proportionally more spent nuclear fuel canisters than high-level radioactive waste canisters, or the spent nuclear fuel canister requires additional measures to control criticality (CRWMS M&O 1997av).

Naval nuclear fuel is designed to operate in a high-temperature, high-pressure environment for many years. Naval nuclear fuel is highly enriched (initially between 93 and 97 percent uranium-235). In addition, to ensure it can withstand battle shock loads, naval nuclear fuel is surrounded by large amounts of structural material made of Zircaloy. There are several different designs for naval spent nuclear fuel, but all have similar materials and mechanical arrangements. The DOE plans to emplace approximately 300 sealed canisters of naval spent nuclear fuel containing approximately

65 metric tons (72 tons) of heavy metal in the repository. The canisters will be transported to the repository by the Naval Nuclear Propulsion Program. The canisters will have a maximum outside diameter of 1.69 m (5.54 ft) and a maximum length of 5.39 m (17.7 ft) (U.S. Department of the Navy Final Environmental Impact Statement for a Container System for the Management of Naval Spent Nuclear Fuel and Department of the Navy, Naval Sea Systems Command draft letter to Russell Dyer, Project Manger, Yucca Mountain Site Characterization Office, October 16, 1997). Each sealed canister will be loaded into a waste package for disposal in the repository; the larger of the two naval spent nuclear fuel waste packages is the heaviest and longest of all waste package designs.

There are 50 metric tons (55 tons) of surplus weapons plutonium to be emplaced in the repository. (DOE 1997c) This material was declared surplus to national defense needs. The current plan is to adopt a dual-track strategy for disposition of this surplus material, as described in the following:

- Fabricate ceramic discs that contain surplus plutonium along with neutron absorber material evenly distributed throughout the ceramic matrix. The ceramic material will resist the leaching of plutonium. Once the plutonium has been immobilized, small cans, each containing a number of the ceramic discs, are placed inside of empty stainless-steel canisters that will later be filled with high-level radioactive waste in the form of borosilicate glass. The canisters for this combination of solidified waste have an outside diameter of 0.61 m (24 in.) and a length of 3 m (9.8 ft), identical to the shorter high-level radioactive waste canister described above in the high-level radioactive waste description. Immobilizing all 50 metric tons of plutonium would require about 1,750 canisters. However, since the canisters would be mostly filled with high-level radioactive waste, only an additional 210 canisters of vitrified high-level radioactive waste would be required. This increase represents the volume of vitrified waste displaced by the small cans of immobilized plutonium.

- Convert a portion of the plutonium into the ceramic form and dispose of as described above and use the remaining plutonium in combination with standard uranium fuel for commercial power reactors. The enrichment of the plutonium-based fuel would have to be decreased (done by blending with other material) to be suitable for commercial applications. This combination is called mixed-oxide fuel and will be a part of the commercial spent nuclear fuel inventory. As such, it is not considered a separate waste form.

The decision of how much of the 50 metric tons will be immobilized and how much will be used as commercial mixed-oxide fuel has not been finalized.

5.1.2 Designs for the Waste Packages

The waste package materials and design have been chosen specifically for a repository host horizon located above the water table in the unsaturated zone. The waste package is designed to contain the nuclear waste for the first several thousands of years, during the time of high thermal output which drives the waste package and repository drift walls well over the boiling point of water. This long-term interval produces a high-temperature, relatively low humidity environment, which greatly diminishes the waste package materials' corrosion potential, thereby enhancing the overall performance of the repository. Many different elements were considered in designing waste packages to meet this objective and to satisfy the requirements presented in Section 3, as follows:

- How will the waste package be designed to assure a subcritical configuration to satisfy the preclosure and postclosure safety criticality requirements defined in Sections 3.1.3, 3.3.1, and 3.3.2 (criticality safety)?

- What are the limits for individual waste package maximum and average heat generation rates to keep the system (repository and waste package) thermal changes at acceptable levels to meet the thermal requirements defined in Section 3.2.1 to maximize repository performance (thermal impact)?

- How strong do the containers have to be under accident conditions to maintain containment to comply with the requirements defined in Sections 3.3.1 and 3.4 (structural integrity)?

- What materials can best withstand the conditions in the proposed repository to produce a long-lived waste package to support the containment and isolation requirements defined in Section 3.1.2 (materials testing)?

- How thick does the container wall have to be to reduce radiation levels that would contribute to corrosion and/or to provide personnel protection (shielding requirements)?

- Can the containers be easily loaded, sealed, and handled?

These design elements must be addressed for every fuel type and waste form that will eventually come to the repository. Criticality safety is addressed by incorporating neutron-absorbing materials in certain packages. High thermal load conditions are partly addressed by using aluminum thermal shunts to enhance heat conduction from the center of the waste package through the inner barrier and to the outer surface of the package. Structural integrity is driven by the need to provide adequate protection of the waste form through design of internal components and is also a major factor in establishing waste package wall thickness. Shielding properties also impact wall thicknesses, while thicknesses and material selection influence the waste package's long-term corrosion performance. Factors such as remote handling, fuel loading, and welding capability affect the geometric design.

No single waste package design will accommodate every fuel type and waste form. Therefore, several variations or design configurations are necessary. These configurations are based on the wastes to be disposed of, along with the predicted safety, long-term performance, and costs. The various designs included in this section include the physical designs of, and the materials used in, the waste package; the bases for selecting the materials; and the general process for fabricating a waste package.

5.1.2.1 Waste Package Designs for Commercial and Noncommercial Waste Forms

This section describes the various waste package designs that have been developed to accommodate specific waste forms.

General Features of Waste Package Designs. Figure 5-2 presents the general features of all the waste package designs. Waste package designs consist of two concentric, cylindrical containment barriers with accompanying lids (a closed cylinder within a closed cylinder) and an internal basket structure, where appropriate (not all designs require a basket). The waste form is arranged in the basket inside the inner cylindrical barrier. Helium is used as a fill gas inside the waste package to help transfer heat from the waste form to the inner barrier and to prevent the waste form from oxidizing.

Containment barriers isolate the waste forms from potentially corrosive agents for as long as possible and retard the migration of radionuclides out of the waste package even after the barriers have been breached. The barriers also provide structural protection from design basis events and eventual collapse of the emplacement drift. To support the waste-containment strategy, the design of the waste package is tailored to take advantage of the conditions of the unsaturated zone of volcanic tuff at Yucca Mountain and to complement the features of the host rock that act as a natural barrier.

As long as the waste packages remain intact, the waste will be contained completely and kept from any contact with the host rock, air, and groundwater. This extended period of containment has several positive results. The radiation source is reduced due to radioactive decay of the wastes. The uranium dioxide contained within the fuel rods

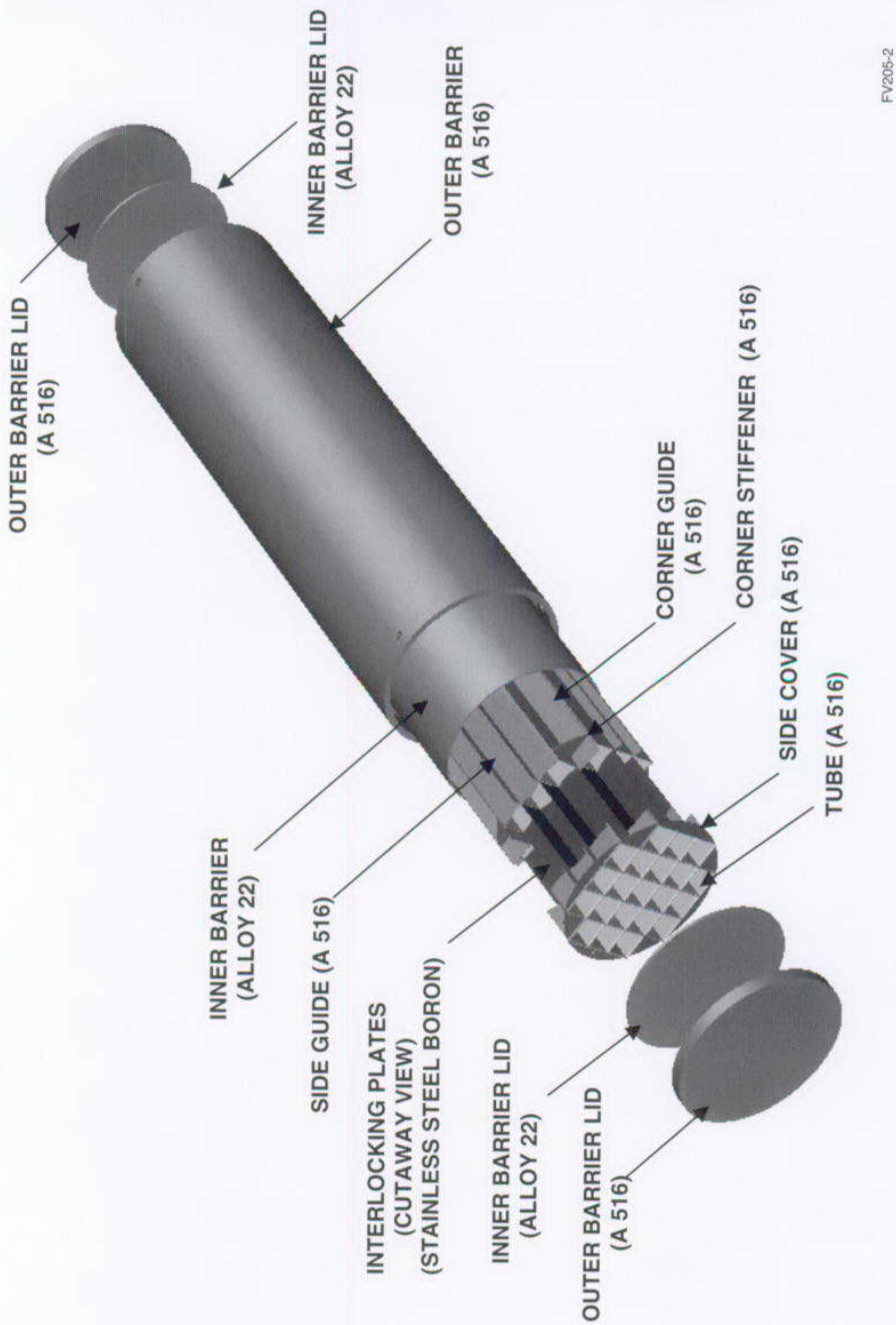


Figure 5-2. Waste Package for 21 Pressurized-Water Reactor Uncanistered Fuel Assemblies

is protected from contact with air during the time it is at a high temperature, which otherwise could accelerate oxidation of the uranium dioxide. In addition, the other waste forms are protected during the initial high-thermal load period. Tests and modeling information already available indicate that waste package containment times exceeding several thousands of years may be achievable.

The inner barrier of the waste package is 20 mm (0.8 in.) thick and made from a solution-annealed, corrosion-resistant material, high-nickel alloy, Alloy 22 or equivalent. For simplification, this material will be denoted as Alloy 22 throughout the rest of the document. The lids for the inner barrier are 25 mm (1.0 in.) thick and made from the same material. The surrounding outer barrier is 100 mm (4.0 in.) thick and made from a corrosion-allowance material ASTM A 516 carbon steel. The lids for the outer barrier are 110 mm (4.3 in.) thick and are also made from ASTM A 516 carbon steel. The outer barrier cylindrical wall extends past the lids forming a skirt at either end of the waste package. This design feature facilitates handling and serves to improve the package's resistance to damage by acting as a shock absorber should the waste package be dropped on end. Due to its greater thickness, the outer barrier initially provides most of the structural protection. However, the inner barrier is designed to be strong enough to protect the waste forms from minor rockfalls long after the outer barrier has corroded away.

The dual-containment barrier design is an example of the defense in depth philosophy of design and provides two lines of defense from corrosion. The combined thickness of the two barriers reduces radiation levels at the surface of the waste package to a level at which radiolysis-enhanced corrosion becomes inconsequential. Under anticipated repository conditions, corrosion of the carbon steel outer barrier will occur evenly and at a predictable rate. The outer barrier may also provide cathodic protection of the inner barrier once the outer barrier is first breached. By the time the outer barrier has been breached, the temperatures inside the waste package will have declined to a level so that any corrosion of the inner barrier would proceed at an extremely slow rate. The inner barrier is expected to continue to provide containment for a much longer period than the time required to breach the outer barrier. Even after the inner barrier has been breached, the waste package will still restrict water flow to the waste form, thus inhibiting the dissolution of radionuclides and their migration out of the waste package.

Waste Packages for Uncanistered Commercial Nuclear Fuel. As described in *Determination of Waste Package Design Configurations* (CRWMS M&O 1997f), the following eight waste packages were designed for uncanistered fuel from commercial power reactors, including mixed-oxide fuel: five for fuel from pressurized-water reactors and three for fuel from boiling-water reactors. As previously stated, the materials and thicknesses of the containment barriers are the same for the all waste package designs.

Baskets inside the waste package are designed to provide structural support to the waste forms during design-basis events or in the event of drift collapse. The different basket designs are tailored to accommodate each of the different waste forms. The baskets are primarily composed of carbon steel and may also include borated stainless steel and/or aluminum. In addition to structural support, the baskets may also provide criticality control and/or facilitate heat transfer, if required for a specific waste form. Criticality is controlled by maintaining the geometric arrangement of the waste form(s), providing neutron absorbing materials (e.g., boron or gadolinium compounds), and displacing water, which is a neutron moderator. The waste package basket, composed primarily of carbon steel, will contribute to moderator displacement upon waste package breach, as the carbon steel corrosion products take up much more space than the original metal, thus reducing the space that could otherwise be occupied by water.

The spent nuclear fuel basket resembles an egg crate put together with interlocking plates. Each basket design is tailored for the size, type, and number of fuel assemblies it must hold. The baskets are composed of interlocking plates, fuel tubes, thermal shunts (where required), structural guides, and absorber rods (where required).

The interlocking plates and fuel tubes set the pattern for how the fuel assemblies will be arranged inside the waste package. The material composition and thickness of the interlocking plates are selected appropriate to the criticality control needs for each waste package design. The interlocking plates are made of either ASTM A 516 grade 70 carbon steel or Neutronit A 978 borated stainless steel, depending on criticality control needs. The thickness of the carbon steel plates ranges from 5 to 10 mm (0.2 to 0.4 in.) for the designs that do not need extra neutron-absorbing material in the plates to control criticality (no-absorber designs) and those that control criticality using separate rods (absorber rod designs). The interlocking plates serve to position the fuel tubes; the fuel tubes provide structural strength to the fuel basket to maintain the fuel geometry during design-basis events.

Some designs control criticality by using interlocking plates made of borated stainless steel. These absorber plate designs use Neutronit A 978 borated stainless steel, with a thickness ranging from 7 to 10 mm (0.3 to 0.4 in.). The corrosion rate for borated stainless-steel plates is low, and the plates tend to corrode by pitting. With this type of corrosion, as the plates corrode, they will remain in place between the fuel assemblies. The borated stainless-steel interlocking plates provide some structural strength to the basket, although no structural credit is taken in the structural analyses. There will be no bends or structural welds on the borated stainless steel plates since these processes tend to stress the material and concentrate boron at grain boundaries, accelerating stress corrosion cracking (a specific type of material degradation).

The fuel tubes are long square open-ended channels that fit inside and support the structure created by the interlocking plates (egg crate) and hold the fuel assemblies in place. The fuel tubes provide structural strength for the basket and the contained fuel assemblies during design basis events. The fuel tubes also aid in conducting heat generated by the fuel. The fuel tubes for each waste package design are made of ASTM A 516 carbon steel that is 5 mm (0.2 in.) thick. As with the interlocking plates, the fuel tubes will further increase moderator displacement as the tubes oxidize following waste package breach.

Where needed, the addition of aluminum thermal shunts provides an efficient means to enhance heat conduction away from the commercial spent nuclear fuel so that the heat can pass through the walls of the waste package. Addition of thermal shunts is a simple and economical method to significantly improve thermal conductance between the center of the waste package and the outer edge of the basket, thereby providing a convenient means to keep the temperature of the cladding within the conservatively specified limits. The shunts are placed alongside the centrally located steel interlocking plates, and along with the steel interlocking plates, become part of the egg crate pattern. The shunts are made of 5 mm (0.2 in.) thick aluminum alloy 6061 T6. Limiting cladding temperatures helps protect the waste form by minimizing damage to the fuel cladding material. Also, the heat radiated from the outer walls of the waste packages to the walls of the emplacement drifts will drive off water in the pores of the host rock and lower the relative humidity inside the drift. Reducing the relative humidity will limit water initially contacting the waste packages and slow the rate of corrosion of the waste package's outer barrier. Current analyses predict relative humidities below 60 percent for hundreds to thousands of years (CRWMS M&O 1995a; Buscheck et al. 1996).

Structural guides hold the basket, or egg crate structure of the interlocking plates and fuel tubes, in place. They extend the full length of the basket and are welded to the inner barrier wall. The waste package structural guides are made of 10-mm (0.4-in.) -thick ASTM A 516 carbon steel.

Absorber rods, similar to reactor control rods, are used sparingly in the waste package design. They are used only in waste packages containing highly reactive fuel assemblies from pressurized-water reactors, where long-term criticality control is required. The absorber rods are made of boron carbide with Zircaloy cladding. Since the cladding on the absorber rods is the same material as that used in most fuel rod cladding, it is expected to have similar corrosion properties and longevity.

Of the five pressurized-water reactor package designs, three are designed to hold 21 assemblies; the other two, 12 assemblies each (see Table 5-1).

Table 5-1. Summary of Waste Package Designs for Uncanistered Fuel from Pressurized-Water Reactors

Number of Assemblies	Design	Inventory (Percent)
21	No Absorber	32
21	Absorber Plates	58
21	Absorber Rods, No Absorber Plates	3
12	No Absorber	5
12	Absorber Plates (Longer Cavity)	2
Total		100

The 21-assembly designs are the same except for the neutron-absorbing material used for criticality control. One design uses interlocking carbon-steel plates with no absorber material; the second (see Figure 5-2) uses borated stainless-steel interlocking plates (absorber plates); and the third has carbon-steel interlocking plates and absorber rods. All three designs use fuel tubes and thermal shunts. The designs for the 12 pressurized-water reactor assemblies differ in length and in absorber material and are designed to hold the higher heat output fuel. One is the same length as the three, 21-assembly designs; the other is longer to accept oversize assemblies and those with higher reactivity. The longer waste package uses borated stainless steel absorber interlocking plates, while the other has no absorber material.

The three waste package designs for boiling-water reactor fuel are all of the same length. Two are designed to hold 44 boiling-water reactor assemblies; and the third to hold 24 assemblies boiling-water reactor assemblies (see Table 5-2).

Table 5-2. Summary of Waste Package Designs for Uncanistered Fuel from Boiling-Water Reactors

Number of Assemblies	Design	Inventory (Percent)
44	No Absorber	25
44	Absorber Plates	74
24	Thick Absorber Plates	1
Total		100

One of the 44 boiling-water reactor assembly packages has no absorber material and the other uses absorber plates (see Figure 5-3). The 24 boiling-water reactor assembly waste package is designed primarily for those fuel assemblies that require additional absorber material and have a higher thermal output. As a result, this design uses absorber plates that are thicker than those in the other waste package designs. All of these waste packages may incorporate thermal shunts.

Waste Packages for Canistered Commercial Nuclear Fuel. Some commercial nuclear fuel may arrive at the repository in canisters suitable for disposal. In such cases, a single canister is placed directly inside a disposal container (waste package). The canister includes a basket to provide the necessary structural, thermal, and criticality performance. There are several canister designs being developed for acceptance at the repository. However, designs that use Boral as the absorber material, which will not provide long-term performance, may require some corrective actions to be accepted for disposal.

There are two waste package designs for canistered fuel, both of which are for boiling-water reactor fuel. One waste package is designed to accept a canister which holds 44 assemblies; the other is designed to accept a canister which holds 24 assemblies. Like the waste packages for uncanistered fuel, these designs use the same dual-barrier concept, materials, and material thicknesses.

Waste Packages for Solidified (Vitrified) High-Level Radioactive Waste. There are three waste package designs for canisters containing vitrified high-level radioactive waste. Two of these designs allow for the co-disposal of a canister of DOE-owned spent nuclear fuel configuration, whereas the third design contains only vitrified waste canisters. The surplus plutonium that has been immobilized will be placed in this third type of waste package.

All of these designs use the same dual-barrier concept with the same materials and material thicknesses previously described, and all are designed to accept five canisters of vitrified waste. The VA reference design does not include a basket

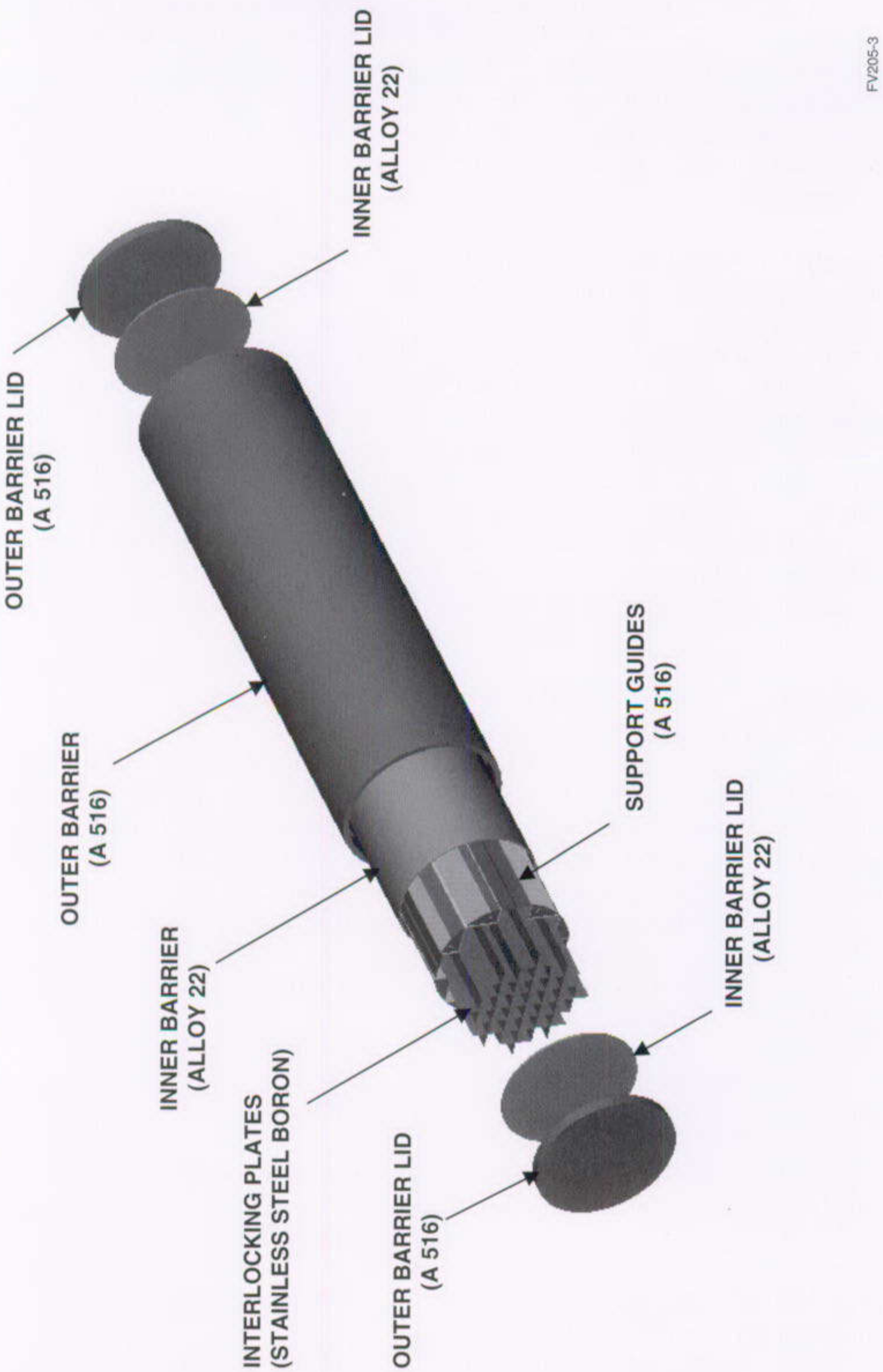


Figure 5-3. Waste Package for 44 Boiling-Water Reactor Uncanistered Fuel Assemblies

to separate the canisters, because special heat transfer and criticality control capabilities are unnecessary. Two of the designs accept a canister of DOE-owned spent nuclear fuel in the center of the waste package (see Figure 5-4). The two designs differ only in length and diameter. The design that does not accept a spent-fuel canister in the center arranges the other five high-level radioactive waste canisters around a guide tube in the center of the waste package.

Waste Packages for Canistered Naval Spent Nuclear Fuel. Naval spent nuclear fuel is to arrive at the repository in canisters suitable for disposal. This fuel is from nuclear powered submarines, surface ships, and training reactors. The canisters will fit one to a waste package similar in design to that for the canistered commercial nuclear fuel. Because there will be two canister sizes, one short and one long, there will be two waste package designs based on the lengths of the canisters. Both designs use the same dual-barrier concept, materials, and material thicknesses as previously described.

Waste Packages for Canistered DOE-Owned Spent Fuel. Although there are many forms of DOE-owned spent nuclear fuel, these waste forms generally are expected to arrive at the repository in canisters suitable for disposal. Some of this fuel will be placed in the center of the waste packages designed for vitrified high-level radioactive waste. However, some fuels within this group may require an additional waste package design either because the spent nuclear fuel canister is too large, or there are proportionally more spent nuclear fuel canisters than high-level radioactive waste canisters, or the spent nuclear fuel canister requires additional criticality control.

Only one of these DOE-owned spent nuclear fuel waste package designs is currently being developed, with the rest being grouped in a miscellaneous category. The N-Reactor fuel waste package being developed will hold four specially designed canisters (multi-canister overpacks). This design uses the same dual-barrier concept previously described.

Waste Packages for DOE-Spent Fuel-Miscellaneous Category. As a consequence of work prioritization, as discussed in this volume, the designs for miscellaneous wastes have not been completed. It is assumed that there will be several designs for canistered DOE-owned spent nuclear fuel; however, these designs are not being defined for the VA reference design.

5.1.2.2 Selecting the Materials for the Waste Packages

The waste package material selection is a multi-step process. First, the waste package is divided into components. Then, the functional requirements for each component are identified. Next, the characteristics of materials that are expected to help meet the requirements are selected. Candidate materials are chosen from the commonly available materials (or, in the case of fill gas, from common gases). The materials are rated against grading scales for these characteristics and suitable weighting factors (level of importance) are applied. Once the materials and alternates have been selected, they are tested, as discussed in Section 5.1.4.

A selection analysis was done for all of the waste package designs. This analysis is documented in detail in *Waste Package Materials Selection Analysis* (CRWMS M&O 1997ap). For the purposes of this analysis, the waste package was considered as comprising the following seven major components:

- Corrosion-allowance barrier
- Corrosion-resistant barrier
- Fill gas
- Interlocking plates
- Fuel tubes
- Structural guides
- Guide tube for high-level radioactive waste glass and/or co-disposed DOE-owned spent nuclear fuel

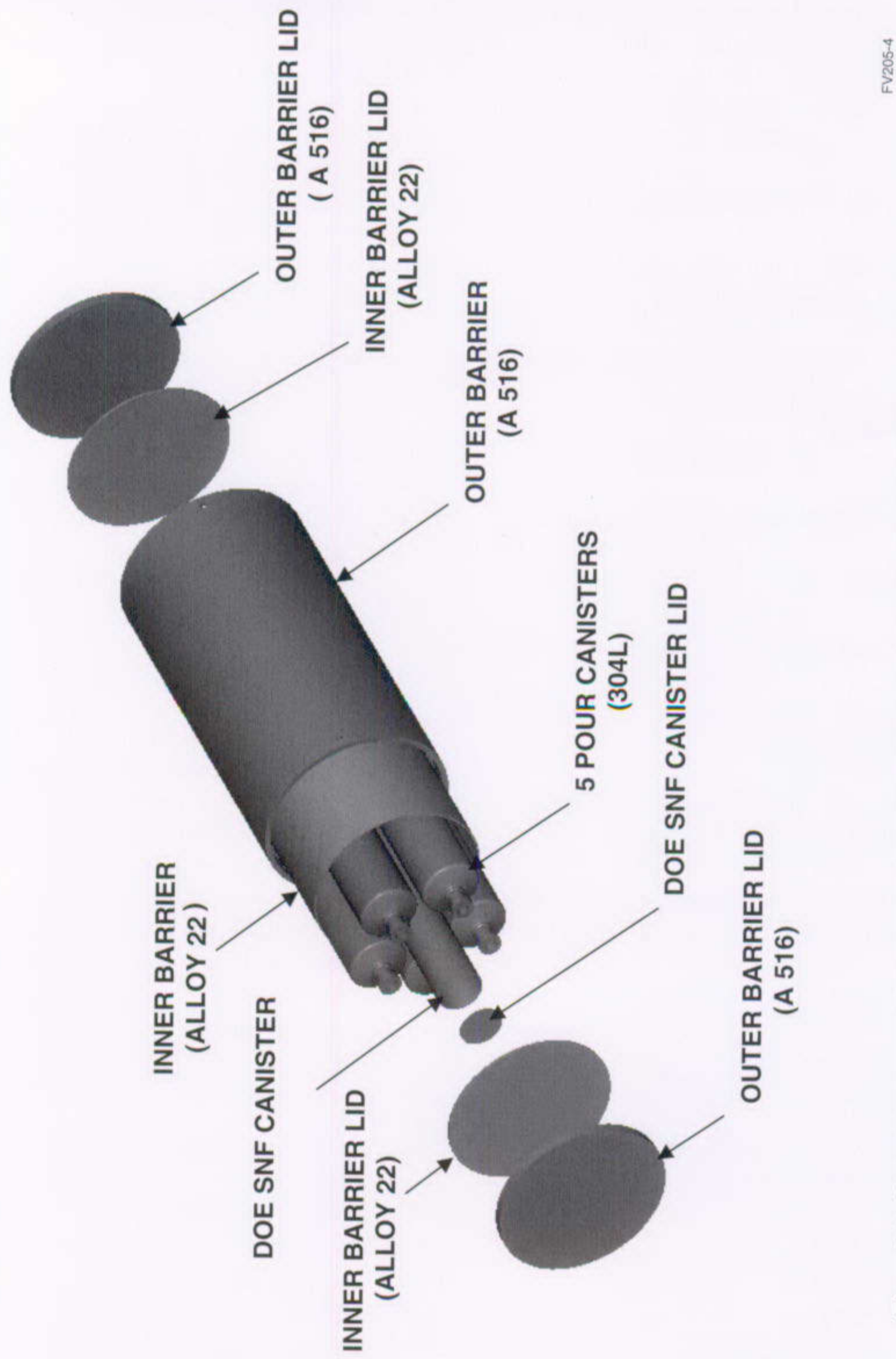


Figure 5-4. Waste Package for Five-Canister, Defense High-Level Radioactive Waste/DOE-Owned Spent Nuclear Fuel Assembly

This analysis was completed by selecting weighting criteria and establishing grading scales. In choosing weighting factors, qualitative arguments were used to justify estimates of the importance of criteria in ensuring that the component will perform its functions. The following nine criteria were identified as contributing to the performance of various components:

- Mechanical performance (strength)
- Chemical performance (resistance to corrosion and microbial attack)
- Predictability of performance (understanding of materials' behavior)
- Compatibility with other materials
- Ease of fabrication using the material
- Cost
- Previous experience (proven track record)
- Thermal performance (heat-distribution characteristics)
- Neutronic performance (criticality and shielding)

Table 5-3 presents the materials selected for waste package components as a result of the selection analysis.

5.1.2.3 Fabricating the Waste Package

This section describes a general method for manufacturing the waste package, as detailed in the *Waste Package Fabrication Process Report* (CRWMS M&O 1997ao). It is very likely that each fabricator, subject to purchaser-procurement-specification restrictions, will adjust the manufacturing and inspection processes as necessary within the limitations of the procurement specifications. The fabrication concept described in this section is for a disposal container designed for uncanistered fuel. The fabrication of the other disposal container designs is expected to be similar, varying only in size and in internal configuration.

Forming the carbon steel outer barrier of rolled and welded plate will require fabrication of two half-length cylinders. Once received by the fabricator, the plate would be laid out to establish the specific length and then cut. The plate would then be heated and rolled to size.

After the heating and rolling process, the cylinder would be adjusted to achieve the required diameter and inside circumference (travel). In adjusting the cylinders, the fabricator must consider the anticipated shrinkage of the longitudinal seam weld. The edges of the longitudinal joint would then be cut to create the proper angle for the weld seam, and cleaned. The seam where the two edges meet would be strutted, or supported, to minimize distortion during welding. The seam would be welded

Table 5-3. Waste Package Component Materials

Component	Material
Dual-barrier design • Corrosion-allowance barrier • Corrosion-resistant barrier	 • Carbon steel, ASTM A 516 grade 55 or 70 • Alloy 22
Fuel tubes	Carbon steel (ASTM A 516 grade 55 or 70)
Interlocking plates	Neutronit A 978 (borated 316 stainless steel)
Waste package fill gas	Helium
Structural guides	Carbon steel (ASTM A 516 grade 55 or 70)
Canister guide for high-level radioactive waste	Carbon steel (ASTM A 516 grade 55 or 70)

ASTM–American Society of Testing and Materials
Note: Where a choice is indicated (ASTM A 516 grade 55 or 70), either of these materials could be used.

using qualified filler material selected to be compatible with the base material.

Upon completion of the long seam weld, the cylinder would be heated in a furnace for about one-half hour at 593 to 621°C (1100 to 1150°F). The struts, or supports, would then be removed and the weld seam tested for flaws. Techniques used for these nondestructive examinations would include radiographic, ultrasonic, and magnetic particle testing. One end of the cylinder would be prepared for welding a circular seam, in preparation for joining two half-length cylinders to form a full-length outer-barrier cylinder. The second cylinder would be prepared in the same manner as the first.

The process for making the inner barrier, or cylinder, would be the same process as that described for the outer barrier cylinder, in which two, half-length cylinders would be fabricated. However, a high-nickel alloy material (Alloy 22) will be used for the inner barrier.

The top and bottom outer barrier lids would be fabricated from 110-mm- (4.3-in.-) thick ASTM A 516 plate. The plate would be laid out, the lid cut to the correct diameter, and edges cleaned to remove the slag and scale left behind from the cutting process. The 25 mm (1.0 in.) thick inner barrier lids would both be cut from a high-nickel alloy plate. The edges of the top and bottom outer- and inner-barrier lids are then machined in preparation for welding to their respective counterparts.

The shrink fit process serves to "line" the outer cylinder with the inner cylinder. This process requires machining the inside surface of the outer cylinder and the outside surface of the inner cylinder, dimensioned so as to produce a slight interference fit at room temperature conditions. After the surfaces have been machined, the outer carbon-steel cylinder would be heated to expand it, following which, the inner cylinder would be inserted. As the outer cylinder cools, it compresses on, or "shrinks around," the inner cylinder, placing the inner cylinder in compression and the outer cylinder in tension.

Once the inner and outer cylindrical assemblies have been assembled into a single unit, the lower lids must be installed and welded in place. Normally, the lid installation would be done with the cylindrical assembly in an inverted vertical position (bottom end up) with the cylinder placed in a pit and the welder suspended above the assembly to weld the lids in place. The welding would then be done in the flat position using the same setup as for the longitudinal seam welds performed during cylinder fabrication. This process includes installation of both lower lids. Commercial weld processes include submerged arc, gas-metal arc, and gas-tungsten arc. After each lid weld is completed, the seam would be examined, as appropriate. Radiographic and ultrasonic examinations ensure that all detectable flaws, regardless of their orientation, are identified, and any identified defects are then corrected. Liquid-dye penetrant and magnetic-particle tests ensure that any imperfections on the surface of the welds are found.

The assembled disposal container (excluding the top lids) would then be heated for three hours at 593 to 621°C (1100 to 1150°F). Following this heat treatment, the accessible weld seams would be re-examined.

Depending on the design configuration of the specific disposal container, the appropriate "basket" components would be assembled and inserted. The carbon steel fuel tubes likely would be welded tubes. The interlocking plates would be simple slotted plates, and no forming or joining would be required. The structural guides would be formed from plate and welded as necessary for subassembly.

Once the cylinder was ready, the internal parts would be attached to the inside of the cylinder. The interior surface of the cylinder would be laid out to establish the location of the structural guides. These guides are installed first and would be welded in place, most likely by manual, gas-tungsten-arc welding. Next, the various interlocking plates would be installed, followed by the fuel tubes. For those disposal container designs which include canister guides, the canister guides would be formed and welded as necessary for subassembly, and then installed in the disposal container.

The inner and outer top lids and the container top inner and outer weld preparations would then be machined for the closure weld (to be performed later at the repository surface facility). The disposal container and the top lids would then be prepared and shipped to the repository for storage until they are ready to be loaded and sealed.

5.1.3 Evaluations to Support Waste Package Design

The information in this section addresses aspects of the NRC Key Technical Issues of Structural Deformation and Seismicity (NRC 1997a), Evolution of Near-Field Environment (NRC 1997b), and Container Life and Source Term (NRC 1998a). This section describes the different mathematical calculations and computer-based models used to develop waste package designs that meet all the necessary criteria for the proposed design and support the waste containment strategy. The section is divided into four topics, each of which affects the performance of the waste packages or the performance of the total repository during operational and postclosure phases. The following topics are covered:

- Criticality analysis and control
- Thermal studies
- Structural evaluations
- Shielding requirements

5.1.3.1 Criticality Analysis and Control

Criticality is the condition of an accumulation of fissionable materials that can sustain a neutron chain reaction. The propensity for such an accumulation to achieve criticality is measured by the effective multiplication factor, k_{eff}; for a critical accumulation, $k_{eff} \geq 1.0$. The objectives of criticality analysis and control are to minimize the occurrence of criticality events and to maintain the risk to the public from such events at an acceptably low value. The criticality potential will vary as a function of time after the waste packages have been emplaced due to the changes caused by radioactive decay, the degradation of the waste packages, and changes in the repository environment. The method for demonstrating criticality control is different for the preclosure and postclosure periods.

The method for analyzing the potential for criticality during the preclosure phase is similar to the one currently used by facilities temporarily storing spent nuclear fuel. The effective neutron multiplication factor (k_{eff}) or criticality potential is calculated for specific events for designs for components containing spent nuclear fuel or waste with fissionable material. The k_{eff} calculated must be 1.0 or greater for a criticality to occur. As indicated in Section 3.1.3, NRC requires that criticality safety be provided by ensuring a 5 percent margin in addition to allowances for any bias in the method used to perform the calculation and the uncertainty in the experiment used to validate the method of calculation (10 CFR 60.131[h]).

The evaluations are performed for events and processes that reflect the range of possibilities in the repository environment. The k_{eff} for each type of waste package design must be evaluated separately including differences in the waste forms. The evaluations that identify what can be loaded into a waste package of a specific design are referred to as loading curve evaluations. These evaluations result in plots that give the acceptable range for material to be placed in the specific waste package designs. The plot considers the initial concentration of uranium-235 (enrichment) and the time and power at which the nuclear fuel operated (burnup). The aspect of burnup is discussed later in this section.

Methods for analyzing postclosure criticality are being developed because neither the current methods for assessing criticality, nor the standard features for controlling criticality consider the changes that occur over long periods of time. The methods being developed are most recently documented in the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997g). The general process for analyzing the potential for criticality of commercial spent nuclear fuel during the postclosure phase is discussed in the following paragraphs.

The waste packages will be designed to preclude criticality unless they are sufficiently degraded to permit the entry of water followed by some degradation of the inner structure (basket) and/or waste form. The variety of possible degradation scenarios is developed as part of the overall method as shown in Figure 5-5. This flow chart indicates that the analysis begins with identifying applicable degradation scenarios based on the following major categories of information: waste form characteristics, waste package design, repository site characteristics, and material degradation characteristics. Figure 5-5 also indicates how the scenarios are traced to identify final configurations with the potential for criticality, and how the configuration specifications are refined by degradation analysis (principally using geochemistry computer programs guided by information on mineralogical changes over geologic time scales). After k_{eff} calculations have identified the configurations that can become critical, including specific parameter ranges, the probabilities of the occurrence of such configurations can be estimated.

The first steps in the method for analyzing "degraded mode" criticality involve assessing the scenarios that can lead to configurations with the potential for criticality. A sequence of degradation stages, starting with a 21 pressurized-water reactor waste package having the insoluble degradation products remaining in the waste package, is shown in Figure 5-6. In this figure, the successive stages of degradation are characterized by increasing amounts of iron oxide, which reflects the progressive corrosion of the steel basket material. The details of the degradation processes are modeled by a geochemistry code. This code is used to compute amounts and chemical compositions of the degradation products, using input parameters to describe the following: chemical components of water flowing into the waste package, and corrosion rates of the steel and the waste forms. Of the degraded stages represented in Figure 5-6, only the last three have the potential for criticality. The earlier stages have sufficient boron remaining in the basket steel to prevent criticality.

Figure 5-7 represents further stages of degradation of the waste package barriers. The nominal assumption is that the inner barrier material will last longer than the assemblies' spacer grids and the cladding on the individual fuel rods (20 mm of Alloy 22, 0.7 mm of Zircaloy). Therefore, only a few rods are expected to remain intact by the time the waste package barriers have completely degraded (see Figure 5-7 B, C). The flat surface shown in each of the diagrams represents the drift floor, and the branched structures shown below the drift floor (in diagrams B, C, D) represent fractures with the potential for accumulating fissionable material and filling with water. If the Zircaloy turns out to corrode so much slower than the Alloy 22 that the assembly spacer grids and cladding last longer than the waste package Alloy 22, then an alternative configuration (not shown in Figure 5-7) could have some undegraded assemblies stacked in the drift after the waste package has completely degraded. Such a configuration could not become critical unless the drift were filled with water to a depth sufficient to cover the assemblies. This is believed to be highly unlikely because the leakage through fractures in the drift floor would be at least as fast as water could enter the drift through fractures in the ceiling (see Section 4.4.4.2 of Volume 3). This configuration will be evaluated for the LA to confirm this assumption.

The annual risk associated with potential criticalities is estimated as the product of the annual frequency (probability) of occurrence multiplied by the annual dose at the boundary of the accessible environment and summed over all possible criticality events. The performance assessment model is used to evaluate the dose rate increment due to the criticality and to compare it with the dose rate from the radionuclide inventory emplaced in the repository as part of the repository's TSPA (see Section 4.4.4 of Volume 3). While the risk attributable to criticality events has been found to be relatively small compared with the performance measures for nominal repository performance, criticality events are nonetheless considered undesirable (CRWMS M&O 1996b). Therefore, the approach also identifies and applies additional control measures that can significantly reduce the probability of criticality. Such a strategy is part of the DOE defense in depth safety strategy (see Section 2.4.1 of Volume 4). The combined strategy of risk analysis and defense in

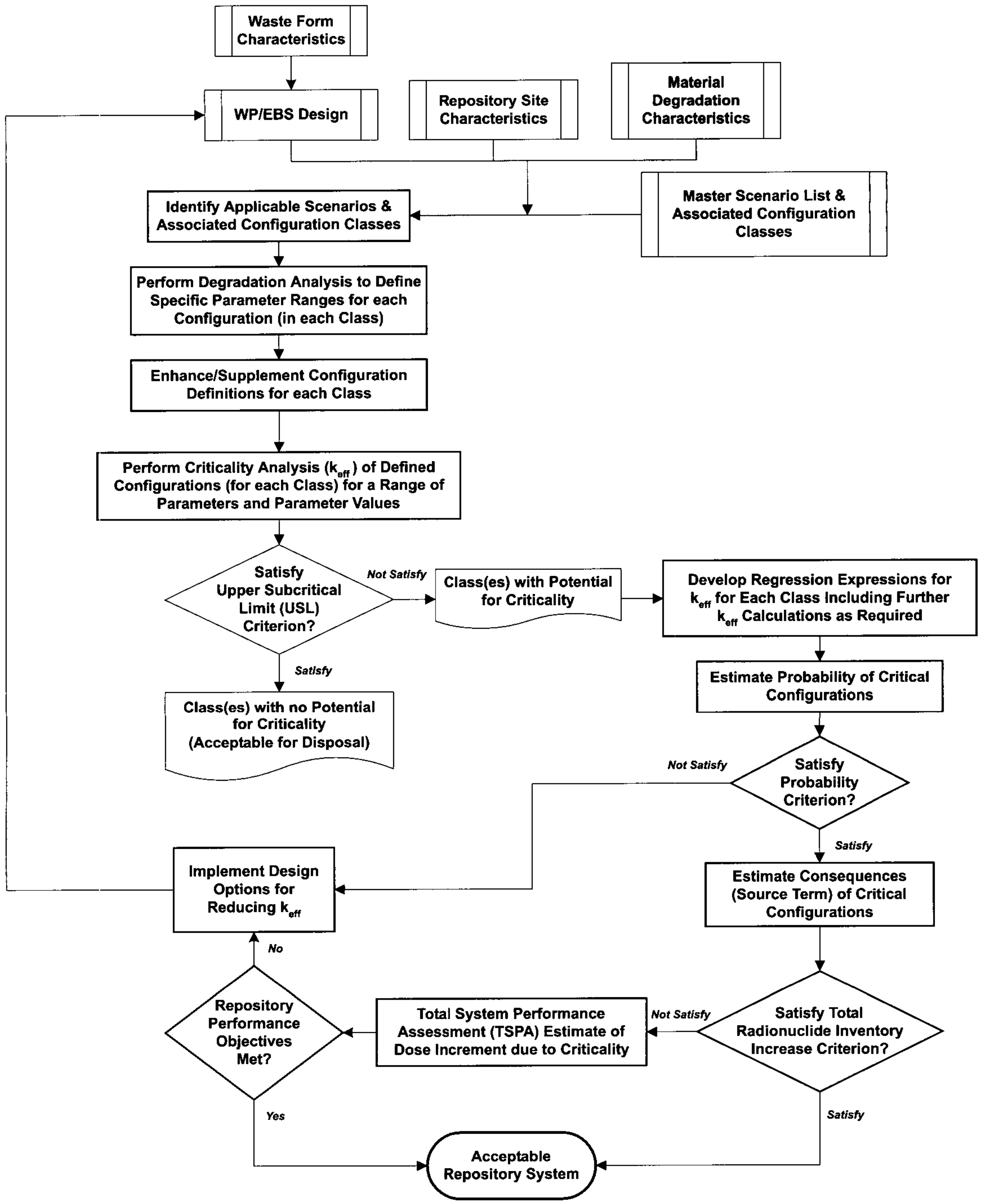


Figure 5-5. Disposal Criticality Analysis Methodology

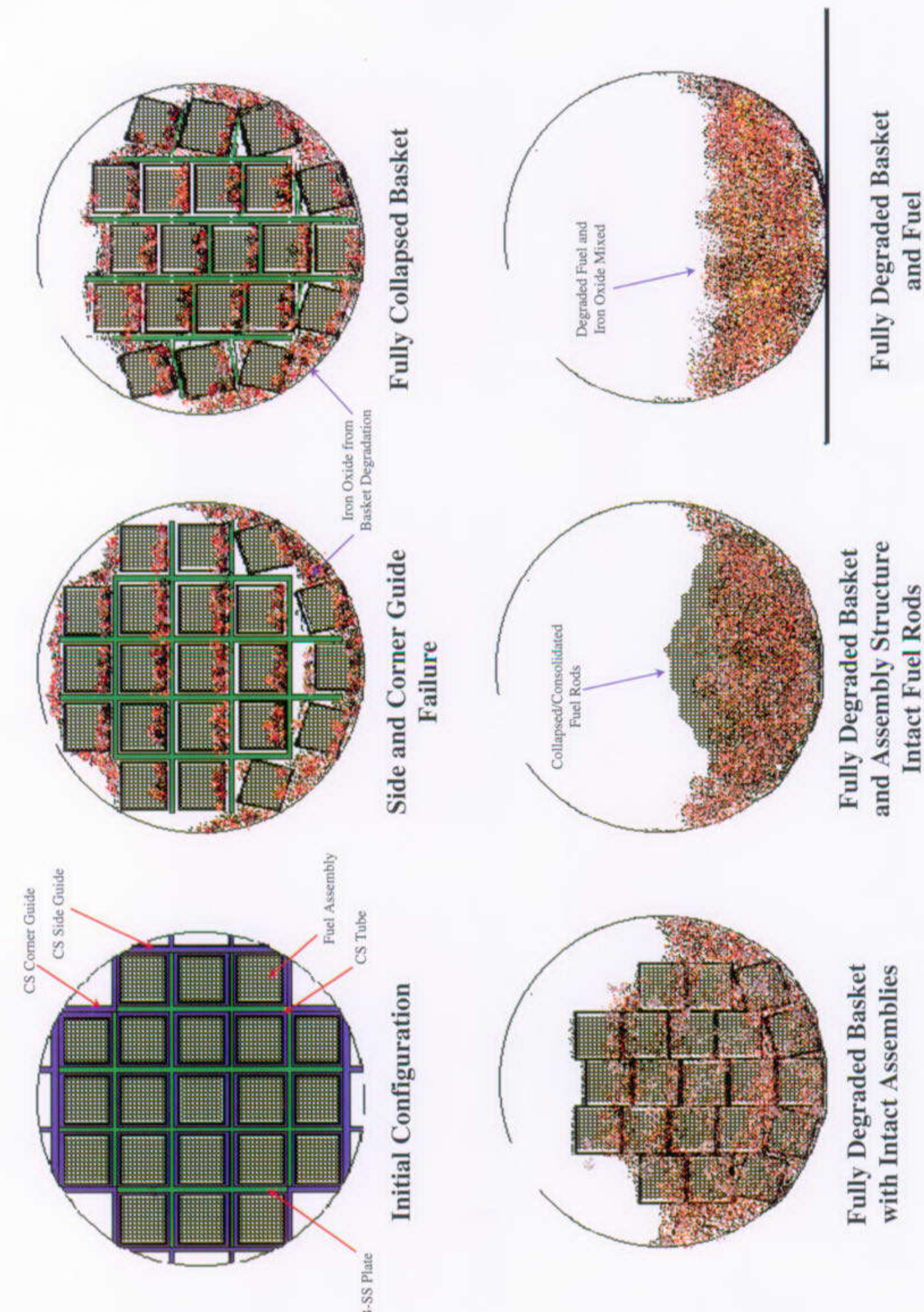


Figure 5-6. Degradation Sequence of Internal Basket Structure and Commercial Spent Nuclear Fuel After Waste Package Breach

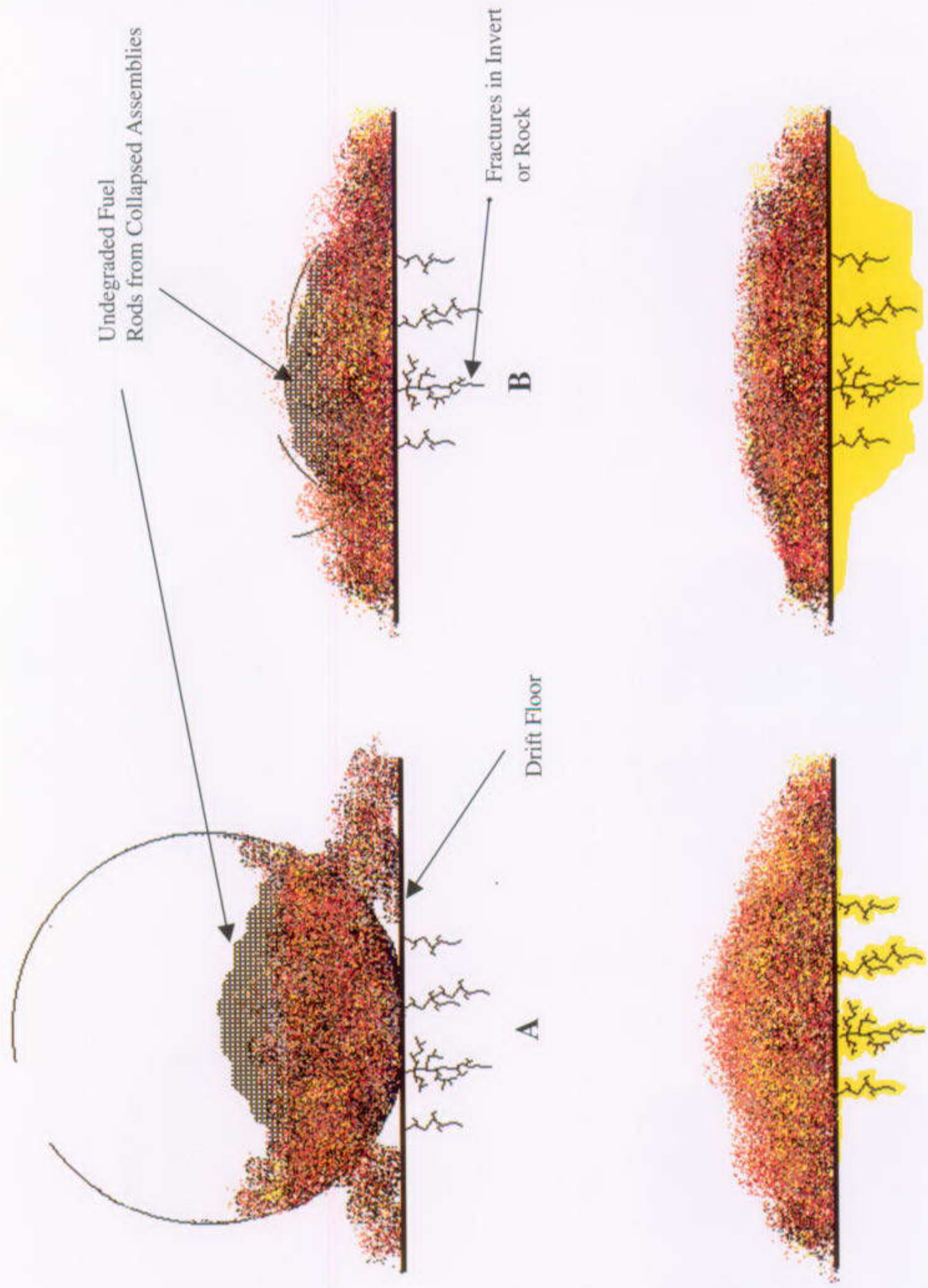


Figure 5-7. Degradation Sequence for Near-Field Environment

depth is referred to as risk-informed, in accordance with recent NRC comments on the subject (NRC interoffice correspondence to L. Joseph Calloy Executive Directory for Operations, "Staff Requirements-COMSECY-96-061-Risk-Informed, Performance-Based Regulation (DSI 12) April 15, 1997). It should be noted, however, that current regulations on disposal criticality (10 CFR 60.131[h]) do not yet reflect this approach. The NRC has stated (61 FR 64260) that these regulations will be revised in the near future to reflect the risk-informed approach for analyzing the potential for criticality during postclosure performance.

The method for evaluating the criticality of commercial spent nuclear fuel will reduce the k_{eff} by accounting for the net depletion of fissionable material and the creation of neutron-absorbing isotopes as a result of fission in an operating reactor. This process is called burnup credit, and the magnitude of the k_{eff} reduction is proportional to the amount of energy that was produced in the fuel during reactor operation (burnup). The burnup credit methodology presented here applies to configurations inside the waste packages. Theoretically, burnup credit can be applied to any spent nuclear fuel, but only commercial nuclear fuel and naval nuclear fuel have the complete burnup histories for the individual assemblies that will be required to approve the method. The use of burnup credit along with a risk-informed method will be presented to NRC for acceptance in the form of a topical report in early fiscal year 1999 (see Section 2.4.1 of Volume 4).

The following design methods may be used separately or in combination to control criticality:

- Limiting the amount of fissionable material
- Adding neutron-absorbing material
- Limiting the amount of moderator (moderator displacement)

The first two are included in the VA reference design; the third is a possible future option. The implementation of these methods is explained in the following paragraphs.

The value of k_{eff} is generally proportional to the amount of fissionable material in the waste package. Therefore, it is possible to control criticality by reducing the capacity of the waste package. However, the required reduction in capacity is generally so large as to make this an extremely inefficient way to control criticality. Capacity reduction is unnecessary for waste forms evaluated thus far.

The addition of supplemental neutron-absorbing materials is a technically acceptable method for controlling criticality. Credit for neutron absorbers is routinely used in reactor and spent nuclear fuel pool analyses. Controlling criticality with a neutron absorber depends on keeping the absorber with the fissionable materials. Therefore, potential mechanisms for loss of the absorber material through long-term processes, such as leaching or preferential corrosion must be considered. Such evaluations will determine the amount of absorber available for controlling criticality during the various phases at the proposed repository.

The amount of moderator in the waste package can be limited by adding a non-contributing filler material (typically metal shot). An effective filler material must be sufficiently insoluble so that it, or its corrosion products, remains in the waste package over the time period of concern. The technique of adding filler is called moderator displacement, and in some circumstances, it may be an efficient way to control criticality.

Development of a loading curve is one way of quantifying the effectiveness of the measures to control criticality. The technique consists of two steps. The first is to calculate the k_{eff} for a set of burnup-enrichment pairs that represent all the expected types of fuel, particularly those expected to give a value of k_{eff} that is near the upper subcritical limit. The upper subcritical limit is the limit below 1.0 that accounts for bias in the computer codes, uncertainty in the experiments used to validate the code, and any additional margin (e.g., the 5 percent mentioned above for preclosure). The second step is to fit this set of values of k_{eff} with a regression for k_{eff} as a function of initial enrichment and burnup. This technique has been applied to the waste package design for 21 uncanistered fuel assemblies from pressurized-water reactors. The results can be shown in terms of three different examples of the regression, with k_{eff} set equal to the upper subcritical limit, under the following different configurations: intact basket with neutron absorber remaining inside the waste package; degraded basket with neutron absorber flushed from the waste package and the iron oxide corrosion product distributed uniformly throughout the waste package; and degraded basket, with iron oxide settled to the lowest 3.5 rows of assemblies. Figure 5-8 shows these three regressions plotted against the distribution of commercial assemblies from pressurized-water reactors. The loading curves divide the fuel into the following two groups: fuel that is subcritical (has a k_{eff} less than the upper subcritical limit in the waste package), represented by dots above and to the left of the curve; and fuel that has a k_{eff} greater than the upper subcritical limit, represented by dots below and to the right of each curve. With this interpretation, the three loading curves determine the fraction of the fuel that has a criticality concern under three different configurations as 3.2 percent, 5.7 percent, and up to 11.2 percent, respectively (CRWMS M&O 1997aq). Applying the third configuration, which is the most conservative, this analysis suggests that additional measures to control criticality may be required for 11.2 percent of the fuel from pressurized-water reactors. (Note that a small portion of such fuel would be placed in a smaller waste package, with capacity for 12 fuel assemblies, see Table 5-1).

The method for controlling criticality has also been applied to waste forms other than commercial spent nuclear fuel. The waste form closest to the commercial spent nuclear fuel is the mixed-oxide fuel (plutonium and uranium), presently planned to be produced from surplus weapons plutonium for use in one or more commercial light water reactors. The criticality of this waste form has been evaluated in the waste package designed for 21 pressurized-water reactor fuel assemblies as used above. This waste package selection facilitates comparison between the commercial low enriched uranium spent nuclear fuel having the greatest criticality potential. For any stage of degradation, and at any time after emplacement, the k_{eff} for such a waste package will be below the subcritical limit. This superior criticality performance of mixed-oxide fuel is due to the fact that all of the fuel will have a reasonably high burnup. However, it should be noted that approximately half of the spent mixed-oxide fuel is expected to have too high a heat generation rate to be put in the reference package for 21 pressurized-water reactor assemblies. These assemblies would be put in packages designed for 12 uncanistered assemblies from pressurized-water reactors. These packages are designed for fuel with high burnup, and therefore, would not require additional material to control criticality beyond that provided by the carbon-steel inner basket. Calculations have shown this to be valid.

Even though the possibility of an internal criticality event occurring inside the waste package is extremely unlikely, the potential effects have been estimated for such configurations. A full explanation of consequence analysis can be found principally in the following two reports: *Probabilistic Criticality Consequence Evaluation* (CRWMS M&O 1997v) and *Criticality Consequence Analysis Involving Intact PWR SNF in a Degraded 21 PWR Assembly Waste Package* (CRWMS M&O 1997c). For a worst case illustration, the criticality event was taken to occur at 15,000 years after emplacement. The following two general types of criticality reaction might occur in the waste package: a slow approach to the critical configuration, or a rapid approach that results in a short high-power pulse.

If there were a slow approach to the critical configuration, the system feedback would control the power output and could remain with k_{eff} at 1.0 for many thousands of years, with a low power output (less than 10 kW). Under such circumstances the only consequence would be an increase in radionuclide inventory in the waste. The results for a 10,000-year criticality, documented in *Probabilistic Criticality Consequence Evaluation* (CRWMS M&O 1997v), showed that except for short times at the end of criticality, the increase in radionuclide

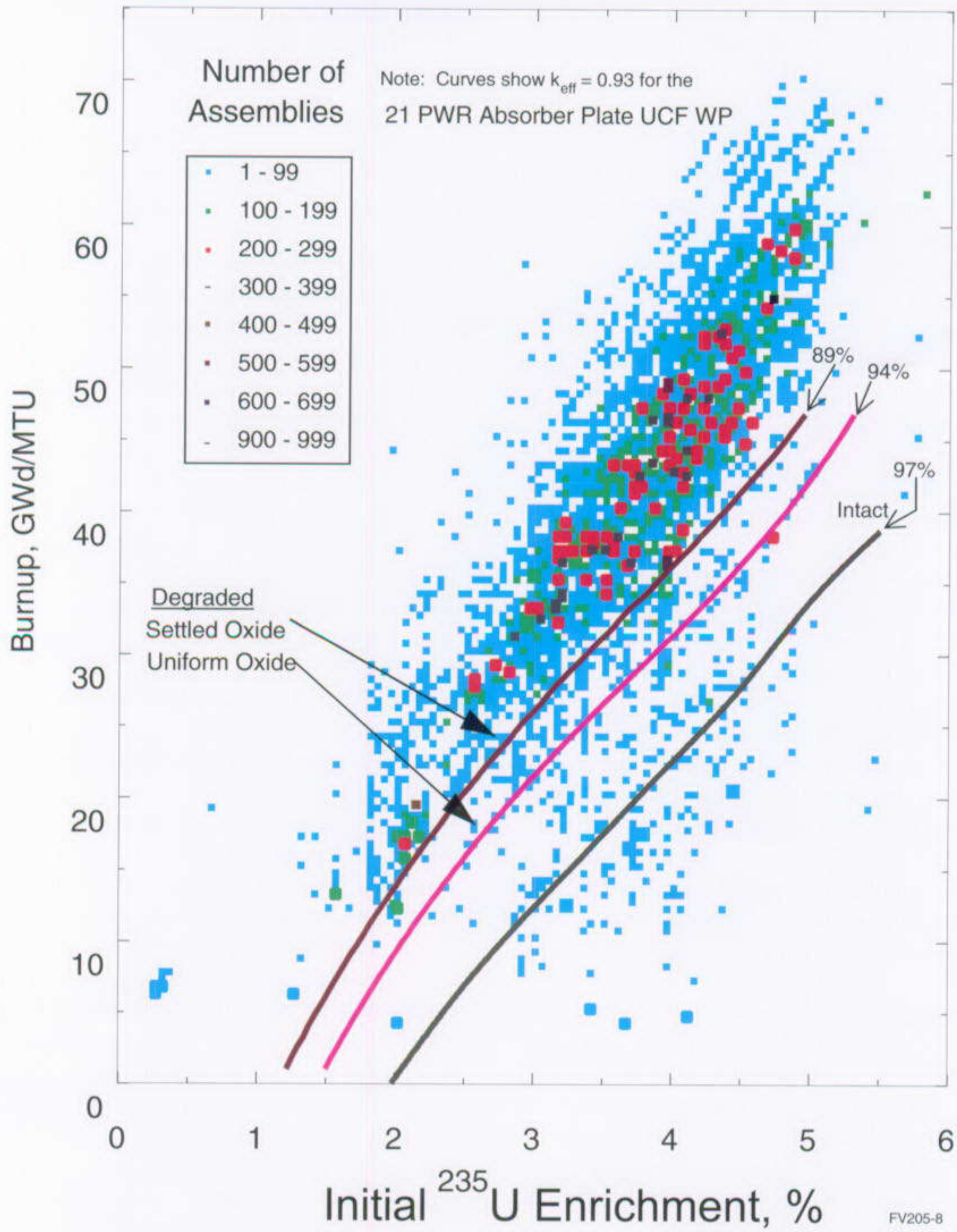


Figure 5-8. Twenty-One Pressurized-Water Reactor Waste Package Loading Curves Plotted Against the Burnup/Enrichment Distribution of the Pressurized-Water Reactor Waste Stream

inventory is less than 10 percent on a single-package basis. If only a few waste packages were to experience criticality, this small percentage increment averaged over the radionuclide inventory of the entire repository would be insignificant.

If there were a rapid approach to criticality, the system feedback would lag the power level rise, and the k_{eff} would increase beyond 1.0, resulting in a short high-power pulse. This transient criticality would be quickly controlled by negative feedback. A transient criticality can achieve significant energy only when the reactivity insertion rate exceeds these negative counterbalancing effects by a significant margin (technical details are described in *Criticality Consequence Analysis Involving Intact PWR SNF in a Degraded 21 PWR Assembly Waste Package* [CRWMS M&O 1997c]). The results of the consequence analysis show that the reference waste package returns to a subcritical configuration, with the fuel temperatures and internal pressures remaining well below levels that could melt fuel or generate more than minor effects on adjacent waste packages. Consequently, criticality events in a waste package would be restricted to localized incidents and would not involve additional waste packages or compromise the repository barriers. The principal impact on the environment external to the waste package experiencing a transient criticality event is the injection of water vapor into the drift environment. This water vapor will quickly condense on cooler, nearby surfaces. This condensation, however, will have no significant impact because it can be no more than the amount of dripping water, which condensed in the waste package in the first place.

In summary, the evaluations for a waste package having 21 uncanistered spent nuclear fuel assemblies from pressurized-water reactors show that the borated stainless steel interlocking plates adequately control criticality for both low-enriched and mixed-oxide fuel. However, a prudent defense in depth strategy would suggest that of the commercial spent nuclear fuel, the pressurized-water reactor spent nuclear fuel with the greatest criticality potential (11.2 percent of the total) be placed in a waste package with the more robust control measure, Zircaloy-clad boron carbide absorber rods.

The criticality methodology has also been applied to commercial nuclear fuel from boiling-water reactors. As with the 21 pressurized-water reactor waste package, the nominal 44 boiling-water reactor waste package will have borated stainless steel plates as the primary criticality control measure. Preliminary analysis indicates that the nominal criticality design will accommodate a larger fraction of the 44 boiling-water reactor spent nuclear fuel than does the nominal design for the 21-assembly waste package for pressurized-water reactors.

Of the variety of DOE-owned spent nuclear fuel, the aluminum-clad matrix fuels from research reactors were found to be among those having the highest potential for criticality. The research reactor specific fuel type selected for initial evaluation was 93.5 percent enriched. The waste package design and supporting analyses for this spent nuclear fuel are described in *Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase II Degraded Codisposal Waste Package Internal Criticality* (CRWMS M&O 1998f). To date, evaluations have been exclusively concerned with internal criticality events. Evaluating the potential for external criticality events is planned for fiscal year 1999, and the results will likely be similar to those for immobilized plutonium, which is also discussed in the second paragraph below.

For the aluminum-clad fuel from research reactors, criticality was evaluated for a range of possible configurations of degraded fuel within a co-disposal waste package concept to identify the most reactive configurations and determine the minimum amount of neutron absorber required to ensure that the subcritical limit was not exceeded. The worst case (highest k_{eff}) configuration would have the degraded, or oxidized, homogenized fuel material in the intact inner basket inside the canister with both fissionable and neutron-absorbing material uniformly distributed throughout the container. The calculation of k_{eff} indicated that approximately 1 kg (2.2 lb) of gadolinium would be required to prevent criticality if stainless steel were used for the basket material; 1.25 kg (2.76 lb) would be required for carbon steel. Stainless steel performs better because it contains some metals that are more neutron absorbing than iron, which is the main constituent of carbon steel. Further, after the stainless steel inner basket has degraded, less than 0.5 kg (1 lb) of gadolinium would be required. If the basket were made of carbon steel, then less than 0.25 (0.5 lb) kg of gadolinium would be required. In this case, carbon steel would perform better than stainless steel because stainless steel typically undergoes localized attack (e.g., pitting). (CRWMS M&O 1998f). Based on the analysis of the aluminum-based fuel, which demonstrates that an acceptable design for disposal can be developed, the other DOE fuel types are also expected to be found suitable for disposal.

Another waste form that contains fissionable material is the ceramic matrix including plutonium (thereby rendering it immobilized), and neutron absorber for criticality control, described in Section 5.1.1. This waste form is designed so that the type and quantity of neutron absorber included are sufficient to prevent criticality in intact configurations, even if the waste package were filled with water. For criticality to occur in or near a waste package of immobilized plutonium waste form, most of the neutron absorber would have to be separated from the fissionable material. This ceramic matrix contains gadolinium and hafnium as neutron absorbers, since they are very insoluble.

The gadolinium in the immobilized plutonium waste form could become soluble, if the solution in the waste package were to become acidic, and if the gadolinium existed as a potentially soluble compound, such as gadolinium oxide. Under such acidic conditions, which are very unlikely, the gadolinium could be flushed from the waste package (CRWMS M&O 1997d, *Degraded Mode Criticality Analysis of Immobilized Plutonium Waste Forms in a Geologic Repository*). Such a possible loss of neutron absorber could be compensated for by limiting the total mass of fissionable material per waste package to less than 50 kg (110 lb). Such a loss of neutron absorber can also be prevented by using an insoluble gadolinium compound or a more insoluble absorber element, such as hafnium.

The method for evaluating the potential for criticality extends to the possibility that external criticality occurs in the environment outside the waste package. The method entails estimating the amount of fissionable material that could be transported by the solution trickling out of the waste package, and estimating the accumulation of fissionable material that can occur by extracting the dissolved material from this solution. This method has been applied to commercial low-enriched spent nuclear fuel, mixed-oxide fuel, and immobilized plutonium. The analyses and results, which vary with the postulated location of the dissolved fissionable material in the external environment, are summarized in the following paragraphs (adapted from the CRWMS M&O 1997k, *Evaluation of the Potential for Deposition of Uranium/Plutonium from Repository Waste Packages*).

There is little likelihood of accumulating fissionable material immediately below the waste package, in the invert, the drift lining, or the first few meters of rock downward for the following reasons that depend on the pH of the dissolving and transporting solution:

- A neutral pH would cause very little deposition of solids of any sort because there is little reaction with the tuff, and virtually no deposition of fissionable material because of its low concentration.

- A low pH (acidic conditions) would cause significant deposition of solids, completely filling the void space at some levels, but very little of that material would be fissionable.

- A high pH would have both the necessary high concentration of fissionable material and the reaction with the tuff to produce an accumulation of fissionable material in the void space of the invert and the fractures of the rock immediately below the waste package. Even in this case, however, the maximum amount of fissionable material that can be deposited in the zone of highest concentration (nearest the waste package) is more than an order of magnitude less than that necessary for criticality.

Further from the emplacement drift, the extensive zeolite deposits beneath the proposed repository site suggest the potential for plutonium or uranium accumulation by adsorption. However, conservative analyses show that the maximum uranium concentration that could be accumulated is 0.166 percent by weight as reported in *Waste Package Probabilistic Criticality Analysis: Summary Report of Evaluations in 1997* (CRWMS M&O 1997aq). If this number were reduced by the 50 percent maximum zeolite concentration in the rock, the density is much too small to form a critical mass, even if the most favorable possible geometry (spherical) and the presence of sufficient water for moderation were assumed.

Several hundred meters from the emplacement drift, there is a possibility (but very unlikely) of localized deposits having the potential for a reducing chemical reaction that could precipitate significant amounts of dissolved uranium. The credibility of such localized deposits may be inferred from natural analogs. This analysis showed that even though epigenetic (materials deposited much later than the host rock) uranium mineral deposits vary in size and grade, a reducing environment is required to cause the precipitation of a uranium mineral. An extensive search of the literature shows six possible types of reducing media: upwelling hydrothermal fluids (water that comes from heated rock), methane, hydrogen sulfide, organic logs, petroleum, and partially oxidized vanadium. Site characterization activities at Yucca Mountain have not shown the presence of more than trace amounts of these media. Thus, the precipitation of uranium minerals by any reduction mechanisms is estimated to have an extremely low probability. Because site characterization activities are still ongoing, new data will be evaluated as it becomes available, to determine if any of these necessary conditions are observed.

The next step in evaluating the potential for external criticality events is to estimate the k_{eff} for those significant accumulations that are representative of the range of possibilities. Such an evaluation is illustrated by considering accumulations of uranium and plutonium in the rock immediately under the waste package, as illustrated by the branched structures extending below the surface in Figure 5-7 B, C, and D. Even using the most conservative assumptions of fracture density, concentrations of fissionable material (uranium or plutonium) in the groundwater, and duration of high concentration outflow from the waste package, the evaluations showed the accumulation of material would be more than an order of magnitude smaller than would be required for a critical mass. This situation would be true even if the optimum concentration of water were available for moderation.

The most important results of the criticality evaluations can be summarized as follows:

- Insoluble neutron absorber either in the basket internal to the waste package (for spent nuclear fuel assemblies) or in the waste form itself (for non-fuel waste forms) will make internal criticality very unlikely.

- Evaluations of external criticality scenarios and configurations have found no physical mechanisms capable of accumulating a critical mass.

- In the unlikely event of a criticality occurrence, the increase in radionuclide inventory and any power transient will have negligible impact on the repository's performance.

5.1.3.2 Thermal Studies

The temperatures in the waste package and the near-field rock are key to isolating radionuclides. Heat reduces the relative humidity in the emplacement drift and therefore slows the corrosion of the metal barriers of the waste package. Further, heat will evaporate water in the rock and thus prevent the transport of dissolved or suspended radioactive materials. However, excessively high temperatures negatively affect the waste package, its contents, and the rock. Therefore, as discussed in Section 3.2.1, maximum (peak) allowable temperatures are set based on criteria for material performance and are specified as design goals for the waste package and engineered barrier system. Peak temperature limits for the waste form(s) and the repository rock are important factors for the thermal design of the waste package. To prevent cladding degradation and prolong the life of the contained waste form inside, peak temperatures for the cladding of commercial spent nuclear fuel are to remain below 350°C (662°F). This limit on temperature is conservative and is designed to protect the metal tubing from cracking or splitting at high temperatures (creep rupture). Maintaining cladding integrity for as long as possible delays contact of air with the waste form and the subsequent onset of waste form degradation. Peak temperatures for high-level radioactive waste glass are to remain below 400°C (752°F) to avoid the temperature at which glass assumes a form that does not effectively isolate the radionuclides (crystallization temperature). And finally, peak rock temperatures are to remain in the following section, 200°C (392°F) at the drift wall and 90°C (194°F) at the zeolite layer, to limit the expansion of the rock at the drift walls and to protect the natural characteristics of the underlying zeolite layer which tends to attract and adsorb certain radionuclides. Such expansion could cause the rock to shift and affect the structural stability of the drifts.

This section discusses how the thermal models are generated and used to estimate the peak temperatures. It also presents how effective thermal conductivity is used to predict peak cladding temperatures in commercial spent nuclear fuel and how the heat output from the waste package is calculated. The effect of the heat generated by the waste form in the waste package (thermal loading) on the surrounding rock and emplacement drifts is also covered. The last subject discussed is the thermal evaluation for the emplaced waste package.

The thermal models for these evaluations were generated using specialized computer software that performs finite-element analyses. The thermal models intentionally provide a conservative estimate of peak temperatures, both within the waste package and in the emplacement drift. The thermal models consider heat transfer by conduction through material and thermal radiation from material surfaces. These models do not consider heat transfer by gaseous convection or thermal-hydrological flow within the rock. Convection is expected to be negligible inside the waste package because the helium fill gas has low buoyancy compared to its conductivity. Convection in the emplacement drift air can also be conservatively neglected since there will be little drift ventilation, and the low conductivity of the rock will result in near-field surface temperatures in a range where thermal radiation becomes the dominant heat-transfer mechanism.

Temperatures in the repository rock will significantly affect water-transport mechanisms. Similarly, liquid-transport and vapor-transport mechanisms will affect the peak temperatures in the repository. However, the effects of water flow always reduce temperatures in the near-field environment. Therefore, temperature results of conduction/radiation models are always conservative compared to thermal-hydrological calculations.

The method for conducting thermal evaluations involves a three-step approach to determine the time-dependent thermal behavior of the waste package. First, a three-dimensional, transient finite-element model of the waste package when it is emplaced provides the history of the surface temperature of the package. Second, this history can be used as a boundary condition in a more detailed evaluation of waste package internals. Finally, the resulting predictions of the temperature of the inner basket from the waste package model provide the boundary for estimating peak temperatures for the fuel cladding.

The thermal environment in the repository will change with time and is affected by the rate of change in the heat produced by the waste packages. Therefore, the thermal evaluation must take into account the time varying heat load of the waste package. This process is called a transient analysis. To determine the interaction between waste packages of varying heat output, a three-dimensional analysis is required. Within the emplacement drift, individual waste packages communicate thermally with each other and the drift wall via thermal radiation, thus resulting in variations of the axial temperature in the drift.

Effective thermal conductivity is used to predict peak cladding temperatures for spent nuclear fuel assemblies. This method, developed in the *Spent Nuclear Fuel Effective Thermal Conductivity Report* (CRWMS M&O 1996i), provides a best estimate of peak cladding temperatures compared to correlations such as "Wooton-Epstein," which produces an undetermined degree of conservatism. Such conservatism often causes a product to be over-designed and more expensive. Rather than modeling the waste package and every fuel rod in every assembly, this method models the fuel assemblies as a smeared (distributed) solid volume with uniform volumetric heat generation. In this model, the smeared properties represent the combined thermal radiation and gaseous transport of heat from the fuel rods to the inner-basket structure.

To determine the appropriate effective thermal conductivities for assemblies from pressurized-water and boiling-water reactors, detailed thermal models (see Figure 5-9) of typical fuel assemblies were developed using a finite-element computer code. Vacuum conditions and fill gases of helium, nitrogen, and argon were evaluated with various rod array sizes (e.g., 8 by 8 assemblies from boiling-water reactors or 15 by 15 assemblies from pressurized-water reactors). For boiling-water reactor fuel, the evaluation considered the effects of channels between fuel assemblies, cladding oxidation, locations of guide tubes and water rods, emissivity variation, and inner-basket temperature gradients. Calculated effective thermal conductivities were found to be highly temperature-dependent due to the contribution of thermal radiation with little dependence on assembly heat output.

Results from this method were compared to those from previous applications using alternate methods and to actual test results from spent-fuel storage casks. The comparisons to experimental data and test calculations indicate that the effective thermal conductivities provide a best estimate of cladding temperatures within a spent-fuel waste package. Further, the effective thermal conductivities were consistent with single-point values previously published by vendors supplying storage casks.

To adequately describe the performance of the waste package, the waste form, and the repository near-field, representative heat outputs for the contents of the waste packages must be determined for each waste package design. In addition to estimates for average heat output for typical packages, design basis heat outputs must be specified to bound maximum temperatures for a given type of waste package. Repository performance is supported both by demonstrating that peak temperatures remain in the following section the goals for the highest or design-basis heat output assemblies or canisters, and by knowing the average, or nominal, repository thermal response.

The analysis titled *Preliminary Design Basis for WP Thermal Analysis* (CRWMS M&O 1997t) developed the average and design-basis heat output as a function of time for each of the waste package designs. Table 5-4 summarizes the initial heat at the time of emplacement for each waste package design described in Section 5.1.2.

Consistent with the licensing of other types of spent nuclear fuel containers, design basis assembly characteristics are defined as part of the design solution for a given waste package design. Each waste package type must be designed and evaluated for the bounding or limiting case fuel assembly that may have a thermal output much higher than average. The design basis fuel represents the limiting characteristics for a fuel assembly that can be loaded into a specific waste package design. Thermal evaluations must consider the variability in the heat output of the waste package-from in the following section average to the hottest case with design-basis assemblies. While all of the waste packages will collectively influence repository temperatures, every waste package design must meet thermal goals.

The first step in the thermal evaluation of the emplaced waste package is to determine the effect of decay heat from the waste package on the repository host rock. The initial heat output from a design basis assembly from a pressurized-water reactor can be more than 75 percent higher than average, which can greatly affect the early temperatures of the repository. However, long-term thermal behavior of the repository rock is primarily a function of the areal mass loading or tons of spent nuclear fuel emplaced per acre. The repository response will be determined more by the integrated heat from the emplaced spent nuclear fuel and less by the initial heat of the individual waste packages. For a given areal mass loading and assuming aver-

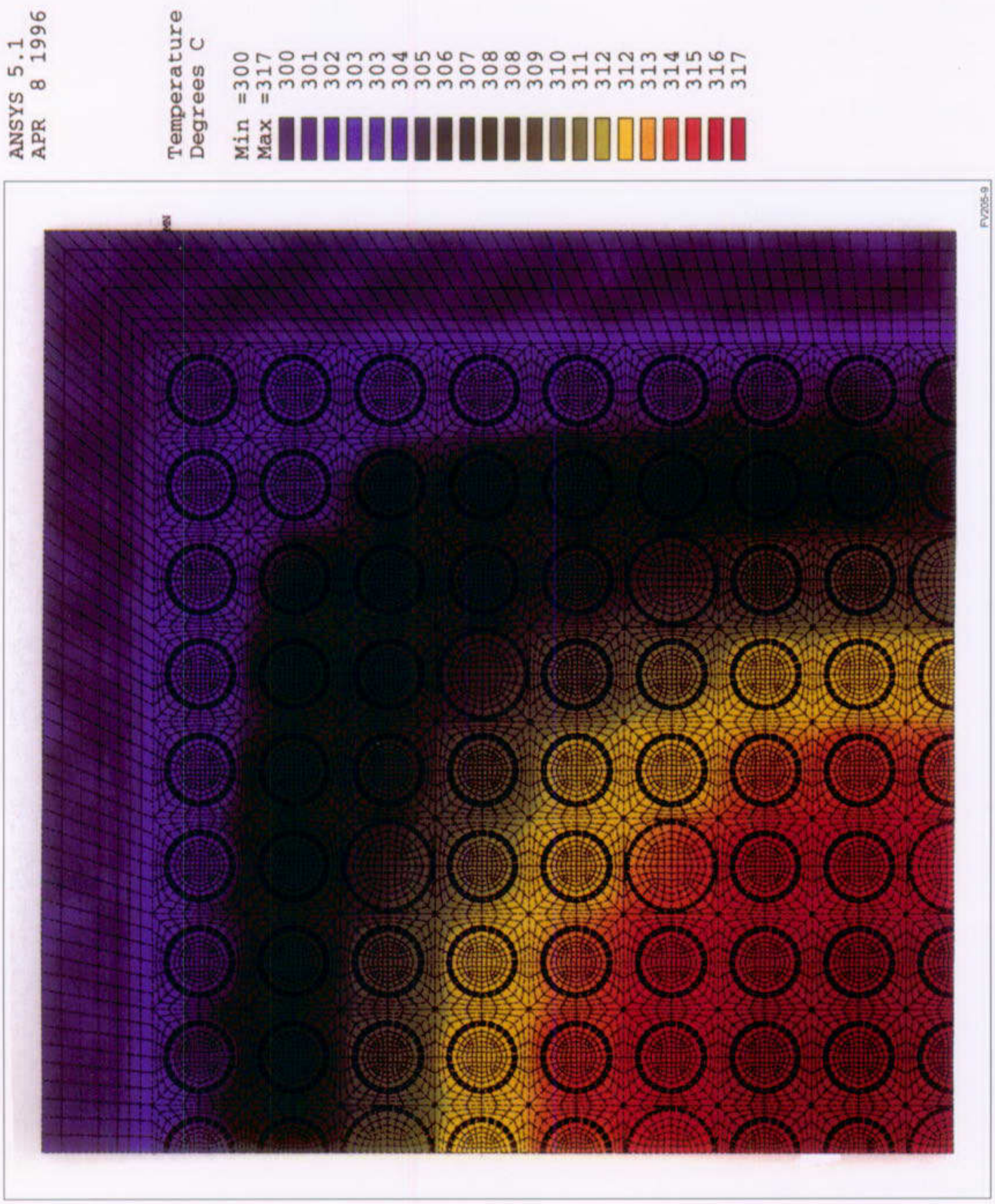


Figure 5-9. 17 by 17 Weld Embrittled Pressurized Water Reactor Spent Nuclear Fuel Assembly - 750 Watts

Table 5-4. Waste Package Heat Output at Emplacement

Waste Package Type	Average, kW	Design Basis, kW
21 PWR – No Absorber	10.27	17.85
21 PWR – Absorber Plates	9.12	17.85
21 PWR – Absorber Rods, No Plates	2.91	17.85
12 PWR – No Absorber	11.23	18.00
12 PWR – Absorber Plates, Long	10.48	18.00
44 BWR – No Absorber	11.63	17.60
44 BWR – Absorber Plates	6.44	17.60
24 BWR – Thick Absorber Plates	0.93	12.48
Canistered Commercial Spent Nuclear Fuel (WESFLEX 44 BWR)	TBD*	TBD
Canistered Commercial Spent Nuclear Fuel (WESFLEX 24 BWR)	TBD	TBD
5-HLW Canister:		
Hanford Site	TBD	4.35
Savannah River Site		3.55
Idaho Environmental and Engineering Laboratory		1.70
West Valley Demonstration Project		1.63
5-HLW/DOE-Owned Spent Nuclear Fuel	TBD	TBD
5-HLW/DOE-Owned Spent Nuclear Fuel – Long	TBD	TBD
Naval Spent Nuclear Fuel – Canistered – Short	4.251	8.010
Naval Spent Nuclear Fuel – Canistered – Long	4.251	8.010
4-Multi-Canister Overpack	TBD	TBD
DOE-Owned Spent Nuclear Fuel – Miscellaneous	TBD	TBD

*To be determined
+kilowatt equals one thousand watts
PWR–pressurized-water reactor
BWR–boiling-water reactor
HLW–high-level radioactive waste

age spent nuclear fuel characteristics, the average temperatures for the host rock can be determined and then applied as the environment for a detailed thermal analysis of the near-field.

To determine the effect of the waste package on repository near-field temperatures, a three-dimensional model of multiple emplaced waste packages was developed. The model was evaluated for a parametric set of thermal loadings, waste package arrangements and rock properties. The calculations will be described in detail in *Multiple WP Emplacement Thermal Response - Suite 1* (CRWMS M&O 1998i).

Figure 5-10 shows the temperature in and around the emplacement drift one year after emplacement for the reference thermal loading of 85 metric tons (94 tons) of uranium per acre. Without significant forced ventilation, temperatures rise above the boiling point of water in less than one year. During the early time period, the temperature varies due to variations in the heat output from the waste packages.

The variations in temperature along the length of the drift decrease with time as heat is redistributed due to thermal radiation and the heat output of the individual packages decreases. The repository rock conducts the heat away slowly, so average drift wall temperatures continue to increase until a peak time of about 50 years after the waste packages have been emplaced. Peak drift wall temperatures near packages with design basis fuel occur earlier due to the high local heat output.

Four years after the waste packages have been emplaced, all sections of the drift wall are above

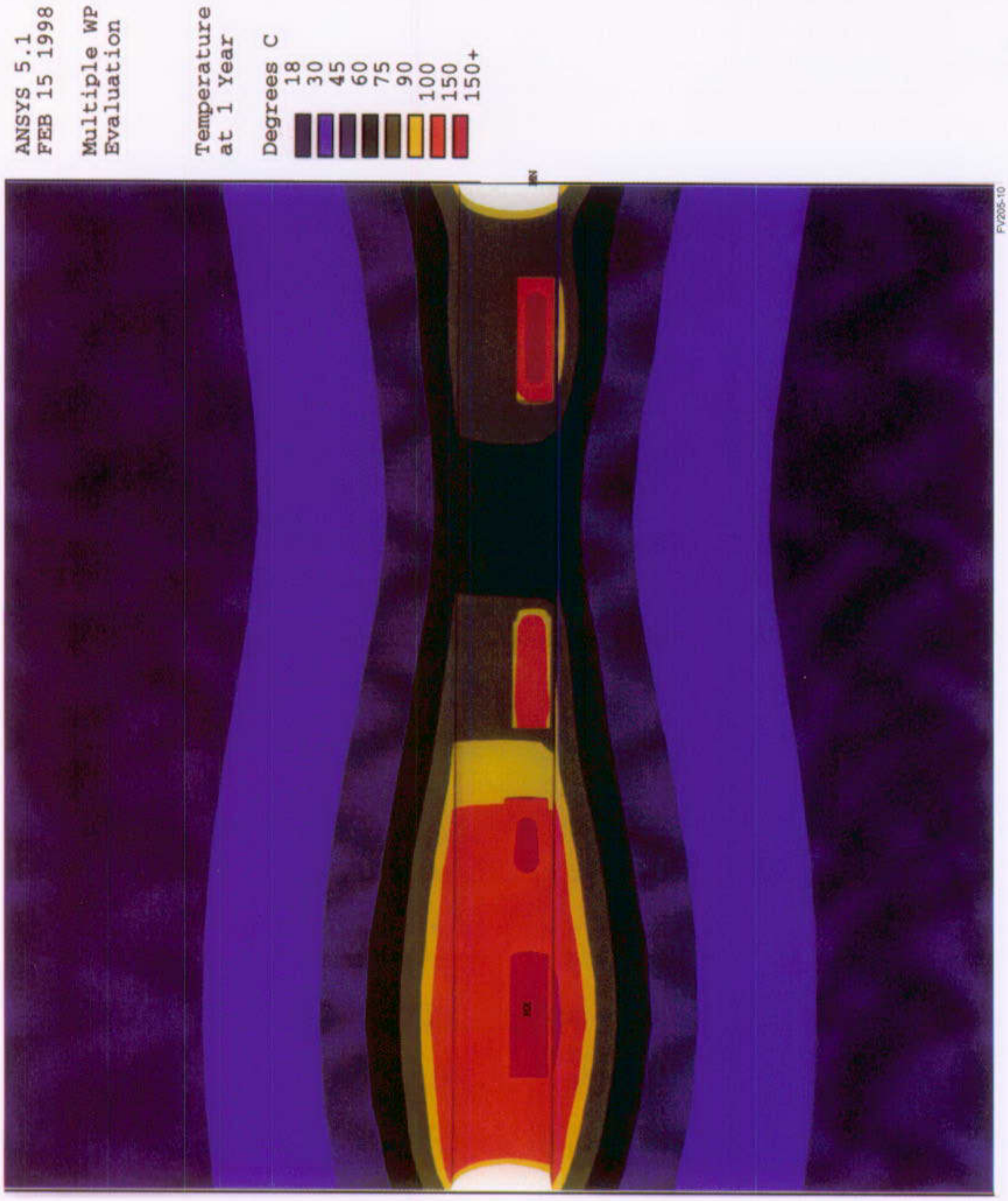


Figure 5-10. Waste Package Emplacement Temperature at One Year

the boiling point of water. After 20 years, the entire drift wall is above 150°C (302° F), while the rock mass between the emplacement drifts is just reaching the boiling point. Depending on the location, average rock temperatures at the repository horizon may stay above the boiling point (100° C, 212° F) for hundreds to thousands of years. The temperature response after 100 years is largely dominated by the design thermal loading, and the variations in temperature between different emplacement locations at later times are small. For the cases analyzed, peak rock temperatures occurred adjacent to a waste package for 21 pressurized-water reactor assemblies with design basis fuel. The most conservative case was selected as the basis for evaluating the internal temperatures of the waste packages. However, peak temperatures of the rock wall are near the limits for the most thermally stressing cases. Average temperatures and peak temperatures near other waste package types were always well within the temperature limits for the rock for all of the cases considered.

To determine the sensitivity of the rock thermal properties on the temperature calculations, different assumed parameters for site properties such as the conductivity of the rock were analyzed. It was concluded that the rock properties significantly impact the magnitude of peak temperature predictions for the emplacement near-field.

Benchmark testing of the thermal models to actual rock thermal behavior cannot be performed until drift-scale thermal tests in the Exploratory Studies Facility are completed. Therefore, it is not known at this time if the most conservative temperature results are too conservative. Several conservative assumptions related to conductivity, ventilation, heat output, thermal hydrology, and waste package arrangement have been compounded in these evaluations.

Peak temperatures inside the waste packages occur much earlier than the peak temperatures in the repository rock. The timing and magnitude of the peak temperature is a function of both the thermal environment in the emplacement drift and the time-dependent heat output for each individual waste package. The analysis of internal temperatures concentrates primarily on evaluating design basis heat output; therefore, the solution results indicate the bounding or worst case, and not the nominal or expected internal temperatures.

Two-dimensional and three-dimensional finite-element thermal models were developed for most of the waste package design types described in Section 5.1.2. Surface temperature histories from the emplacement-scale evaluations provide the boundary condition for transient thermal analyses of the waste forms and the internal structures of the waste package. For commercial nuclear fuel, a two-dimensional cross-section model is adequate to determine peak temperatures. However, for high-level radioactive waste glass, both two- and three-dimensional models were used.

As depicted in Figure 5-11, the peak temperature of the fuel cladding in the 21 PWR waste package is 332°C (630°F) and occurs two years after emplacement—much sooner than the peak temperatures occur in the surrounding rock (*Thermal Evaluation of Preliminary 21 PWR UCF Design - Internal Structures* CRWMS M&O 1997at). The magnitude of the peak temperature results from the combination of temperature drop from center-to-edge of the waste package and the environment temperature in the emplacement drift. The internal temperature drop depends on the heat load and the material design of the inner basket, while the environment is influenced by all of the factors discussed in the previous section. Therefore, a change in rock properties, for example, will directly affect the peak cladding temperature. A less conservative rock conductivity would reduce drift temperatures. This effect would directly result in a reduction in peak cladding temperature. Maintaining the integrity of the cladding prolongs the life of a barrier between the waste form and any water or oxygen that may enter a breached waste package. The peak cladding temperature will have dropped significantly after the first 100 years. Section 5.5.2 of Volume 3 discusses the importance of cladding in repository performance.

Several design options for each type of waste package have been considered. For example, using the aluminum plate (thermal shunt) will lower peak

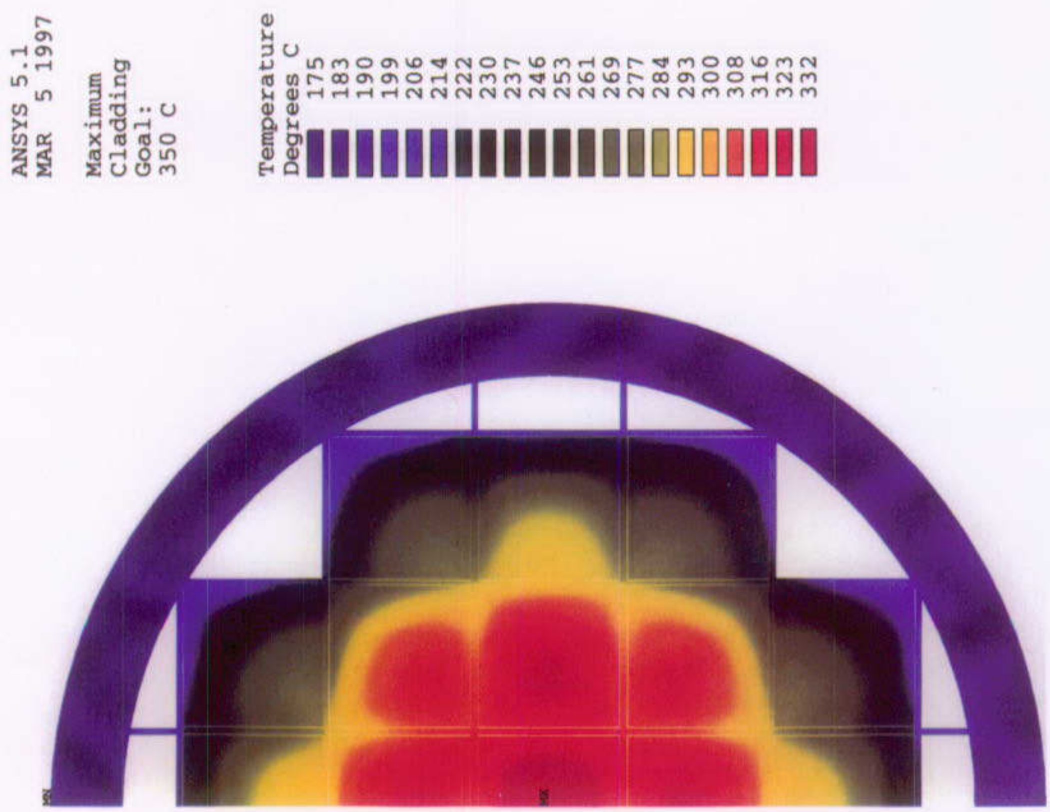


Figure 5-11. Twenty-One Pressurized-Water Reactor/Waste Package at Time of Peak Temperature (2 Years)

cladding temperatures in the waste package design for 21 assemblies from pressurized-water reactors. Similar calculations were performed for each waste package design.

In summary, the results for the thermal evaluation indicate that maximum temperature limits can be met for each of the waste package designs. However, in some cases that use the most conservative estimates, the peak temperatures may be at, or near, the maximum allowable temperatures. This work addresses aspects of the subissue "effects of coupled processes on waste package lifetime" associated with the Evolution of the Near-Field Environment Key Technical Issue (NRC 1997b). Status information about this subissue is summarized in Volume 4, Section 4.3.3.3. (NRC 1997b).

5.1.3.3 Structural Evaluations

To demonstrate compliance to preclosure radiological goals and objectives (Section 3.4), all of the designs for the waste packages are evaluated to see how they will react under certain off-normal or accident conditions, or design basis events (CRWMS M&O 1997au). The following three main types of evaluations assess what happens: if the waste package were dropped during handling, either at the surface or underground; if the waste package were to tip over; and if rocks were to fall onto waste packages lying horizontally in the drift. The designs have also been evaluated for their performance under other unusual conditions, such as the impact of an object released under force, or pressure. Evaluations also determine how the different designs behave when they are pressurized, or exposed to heat, or are lifted during handling. This section briefly describes the tests and the criteria that must be met.

Some of the design basis events for handling involve dropping the waste package. For example, it is possible that the waste package could be dropped while being held vertically from a height as high as 2.0 m (6.6 ft). A drop while the package is being held horizontally could be from as high as 2.4 m (7.9 ft). These analyses are performed to verify that the waste package does not crack or break open, and that the inner basket does not deform so that the fuel assemblies are crushed or pinned inside the basket such that they could not be easily removed.

A finite-element computer code has been used to evaluate the standard waste package designs for these types of drops. The analyses are performed on three-dimensional geometries and calculate the immediate dynamic impact, which includes the elastic-plastic properties of the material and nonlinear deformation. The drop evaluations are performed for the following orientations given in degrees from vertical: 0, 30, 60, and 90. Drops are also performed with the center of gravity over the corner of the package. The drops for 90 degrees from vertical are also run at interlocking plate orientations in the inner basket of 90 degrees and 45 degrees. Evaluating these drops determines the angle, or orientation, that causes the most damage to the basket. The analysis for the 45-degree basket orientation is documented in *Horizontal Drop of the 21-PWR Uncanistered Fuel Waste Package on Unyielding Surface with 45 Degree Basket Orientation* (CRWMS M&O 1998g) and the computer results are presented in Figure 5-12. Shown in the figure are stress contours, identifying the locations where stresses are high or indicating the weakest points in the design of the basket and the waste package barriers based on a drop design basis event. The results given in the horizontal drop analysis show that the maximum membrane plus bending stress on the inner barrier is 417 MPa. A comparison of this value with the allowable stress ($0.9S_u$), 621 MPa, reveals that the inner barrier design meets the American Society of Mechanical Engineers Code requirements.

The stresses determined in these evaluations, as well as the two other types of evaluations described in the following sections, are compared with allowable limits for stress intensity from the 1995 *ASME Boiler and Pressure Vessel Code*, Section III, Subsections NB and NG, and Appendix F (ASME 1995). The material design stress, yield strength, and tensile strength are taken from the 1995 ASME Boiler and Pressure Vessel Code, Section II.

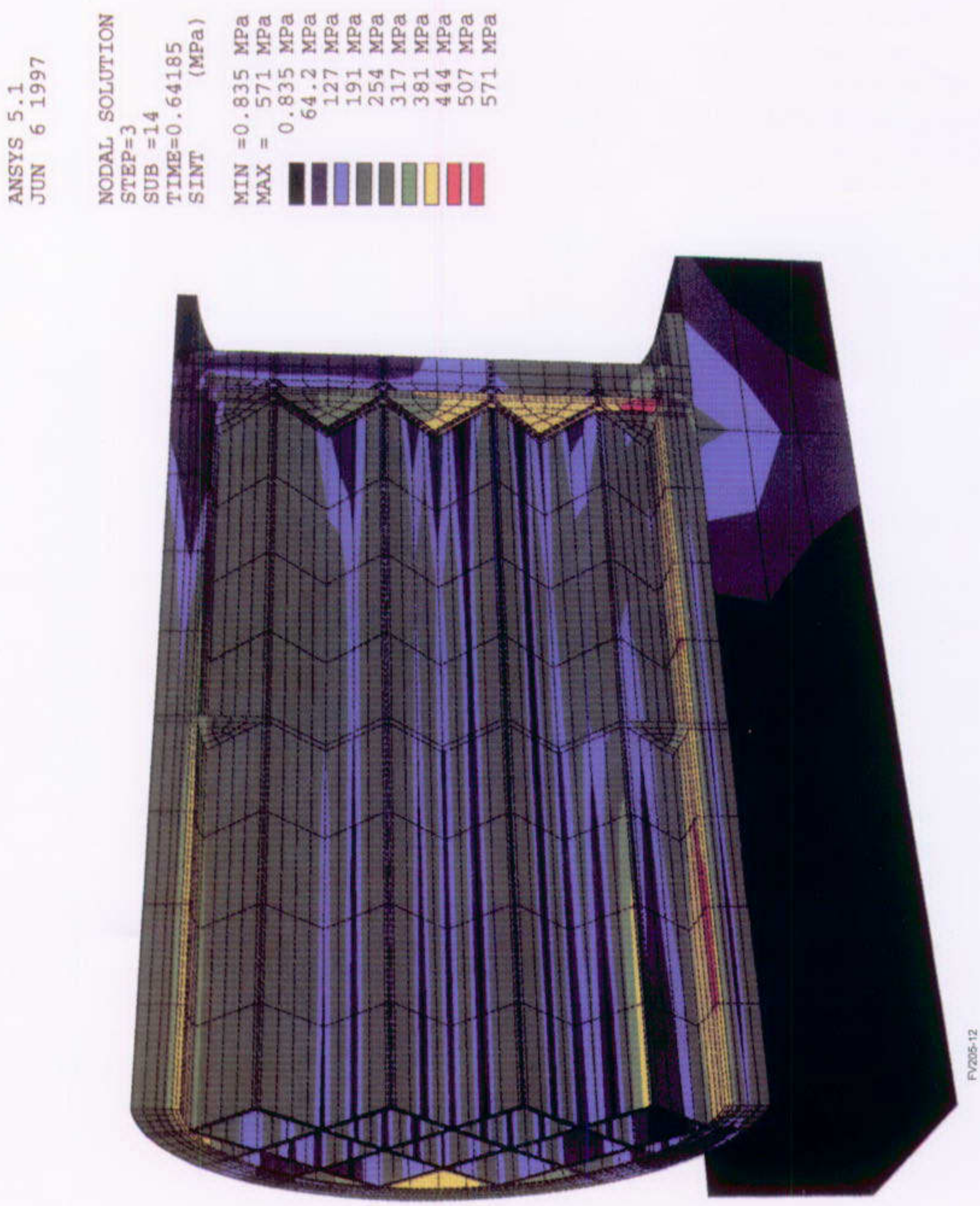


Figure 5-12. Two-Meter Drop of 21 Pressurized-Water Reactor Waste Package, Horizontal-45 Degree Basket

Another design-basis handling event is the waste package tipover. A waste package can tip over as a result of being dropped, or from seismic activity. These evaluations, performed using the same analysis code, have the same objectives as those given for the evaluations described for dropping the waste packages. However, the evaluations are performed only for the orientation of the drop that would cause the most damage to the inner basket. The analysis is documented in the *UCF WP Tipover Analysis* (CRWMS M&O 1998l) and the computer results are presented in Figure 5-13. Shown in the figure are stress contours identifying the locations where stresses are high or indicating the weakest points in the design of the basket and the waste package barriers based on a tip over design basis event. The results given in the *UCF WP Tipover Analysis* show that the maximum membrane plus bending stress on the inner barrier is 456 MPa. A comparison of this value with the allowable stress ($0.9S_u$), 621 MPa, reveals that the inner barrier design meets the American Society of Mechanical Engineers Code requirements.

It is also possible for rocks or a drift lining to fall onto a waste package after it has been emplaced. Therefore, these events must be evaluated both for fully intact waste packages that have been recently emplaced in the repository and also for packages that have begun corroding. The rockfall evaluations determine the sizes of the rock, or critical rock mass, that can cause the package to crack or open. Corroded packages are represented as packages with thinned barriers. In these analyses, the critical rock mass to breach a package is plotted versus the thickness of the barrier (walls of the waste package) remaining. These analyses are also performed using the computer program described above. However, the main criteria for these evaluations are to avoid breaching the barriers and to maintain the geometry of the material used to prevent internal criticality. Pinning of the fuel is not an issue because after the repository has been closed, as the waste package and fuel no longer have to be retrievable. The computer results are illustrated in Figure 5-14. Shown in the figure are stress contours in the waste package barrier caused by the impact of a rock. The following three separate analyses may be referred to for more detail: *Rock Size Required to Cause a Through Crack in Containment Barriers* (CRWMS M&O 1996h); *Rock Size Required to Breach Barriers at Different Corrosion Levels* (CRWMS M&O 1996g); and *Emplaced Waste Package Structural Capability Through Time Report* (CRWMS M&O 1996a). It has been concluded from these analyses that when there is no degradation of the barriers, the waste package is capable of withstanding the fall of a 38,000 kg (83,755 lbs) rock, which is considerably more than the design basis rock mass (25,000 kg, or 55,115 lbs).

As mentioned in the introduction to this section, evaluations have been performed and are documented in the *Missile Impact Analysis of UCF Waste Package* (CRWMS M&O 1997p) to determine if the waste package can survive impacts of objects or missiles, which result from the failure of pressurized components. This design basis event comes from the design basis event document. The evaluation performed was based on the impact of a 0.5 kg (1 lb) missile 0.01 m (0.4 in.) in diameter propelled at 5.7 m/s (18.7 ft/s). The calculations showed that this event was inconsequential in terms of stress on the waste package barriers.

Evaluations reported in the *UCF WP Handling and Lifting Analysis* (CRWMS M&O 1997am) have also determined if waste packages can be lifted vertically and horizontally as designed. These analyses were performed as static analyses, with stresses being compared to allowables determined from the 1995 *ASME Boiler and Pressure Vessel Code*, Section III, Subsection NB (ASME 1995). The material design stress, yield strength, and tensile strength are taken from the 1995 *ASME Boiler and Pressure Vessel Code*, Section II (ASME 1995). This analysis was performed for a package weighing 70,000 kg (154,322 lb). At the time that the analysis was performed, this weight addressed the heaviest package expected at the repository. The waste package design for naval spent nuclear fuel, however, exceeds 70,000 kg, and therefore, an analysis to support the heavier weight will be performed. Results showed that the 70,000 kg package could be lifted both ways.

Static load, thermal stress, and internal pressurization were analyzed together in the *UCF Static*

Figure 5-13. Waste Package Tipover Analysis for 21 Pressurized-Water Reactor Uncanistered Fuel Assemblies

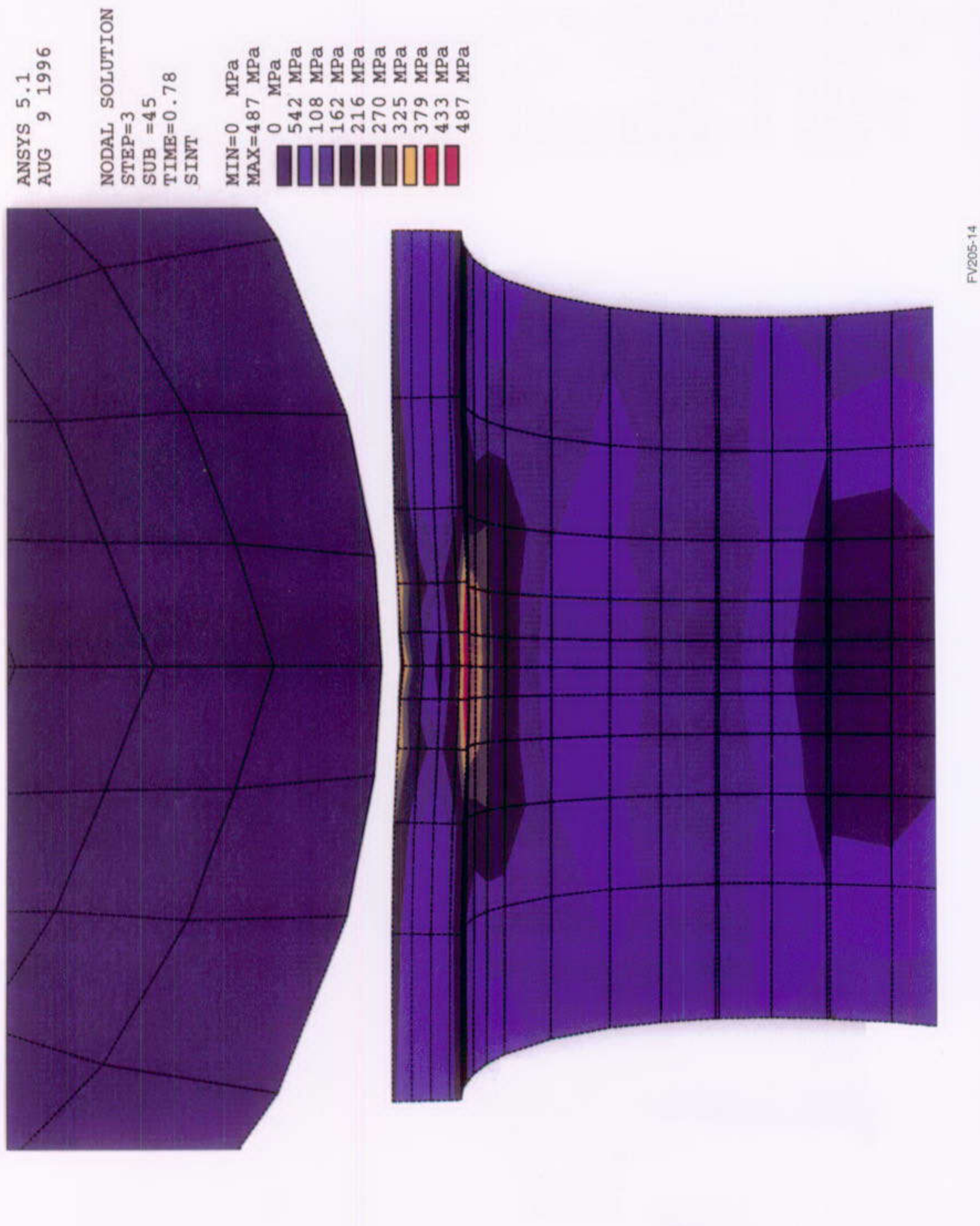


Figure 5-14. Waste Package Rockfall Analysis for 21 Pressurized-Water Reactor Uncanistered Fuel Assemblies

Load, Thermal Expansion Loads, and Internal Pressure Analysis (CRWMS M&O 1997al). Both the stresses in the barriers and the internal pressurization were very low. Even with the static load added in, the barrier stresses were still inconsequential. The internal basket components were sized not to be stressed by thermal loads.

Static structural analyses were performed through time as the barriers and basket components corrode. Corrosion was modeled as thinning of the components. These evaluations, documented in the *Static Structural Analyses of Waste Packages in Degraded States* (CRWMS M&O 1996j), showed that the waste package can withstand static loads, including those imposed by backfill, until the entire outer barrier and more than half of the inner barrier have corroded away, which will take thousands of years. Internal basket components will be able to maintain their geometry until they have corroded halfway through their thickness. The inner basket would not begin to corrode until the barriers of the waste packages have been breached.

The work described in this section addresses aspects of the subissue "What are the effects of materials stability and mechanical failure on the lifetime of the containers and the release of radionuclides to the near-field environment?" This subissue is associated with the Container Life and Source Term Key Technical Issue (NRC 1998a). Status information about this subissue is summarized in Volume 4, Section 4.3.3.4.

5.1.3.4 Shielding Requirements

Shielding is material that protects personnel, equipment, and materials from damage due to the radiation. Sources of ionizing radiation, such as spent nuclear fuel and other high-level radioactive waste, normally require shielding. Radiation shielding evaluations are used to determine how much shielding is required or what the radiation dose rate is for a specific source.

During normal operations at the repository, the waste packages loaded with spent nuclear fuel will be handled remotely from within shielded areas so that personnel will not be exposed to harmful radiation levels. In this instance, shielding evaluations are performed to protect material and equipment.

Waste packages must be thick enough to preclude radiolysis enhanced corrosion. The barriers must be able to reduce radiation levels at the waste package surface such that they are too low to enhance corrosion under aqueous conditions. It appears that the radiolytically induced corrosion is insignificant due to the very low initial gamma dose on the surface of the waste package and the fact that the gamma dose declines by about a factor of 8 to 10 each 100 years. Thus, when the packages could be contacted by water after the thermal pulse has passed, the dose on the surface will be at very low levels and below the threshold for radiolytically induced corrosion.

Shielding evaluations are done for equipment to determine the amount of protection a piece of equipment needs to function properly at a given location for a certain period of time.

Various pieces of monitoring and control equipment will be relatively close to the waste packages. The results of the shielding evaluations will be used to determine the amount of protection a piece of equipment needs to function properly at a given location for a certain period of time.

5.1.4 Evaluations of Waste Package Materials and Waste Forms

The overall strategy for waste containment and isolation is based on a defense in depth approach in which the waste package is a key component. The first part of the strategy is to contain the waste in waste packages for an acceptable period of time (containment) and the second part, isolation, entails ensuring that the release rates are acceptably low (controlled release).

The waste package inner and outer barriers provide the dual-barrier containment system for the waste package. The materials selected for the barriers must be evaluated for their expected performance in a broad range of possible repository environments. The performance of the waste package is estimated using a waste package degradation model. It provides a quantitative analysis of the initial waste package failure and its degradation through time. Detailed results of the waste package degradation model are provided in Section 3.4 of Volume 3 and are summarized in this section.

The first penetration occurs in the waste package corrosion-allowance outer barrier at about 1,000 years and the corrosion process continues until the outer barrier fails for all waste packages at approximately 6,000 years. The base case analyses incorporate both dripping and non-dripping conditions. The waste packages that are dripped on have a higher percentage of failures for the corrosion-resistant inner barriers and fail much earlier than the waste packages that are only exposed to humid air conditions within the drift. All waste packages that are dripped on fail by 1 million years.

The waste forms will be initially contained in the waste packages. After degradation and failure of the waste packages, spent nuclear fuel assemblies and high-level radioactive waste glass will be exposed to the drift environment. After waste package failure, radionuclides are not available for release and transport until the combination of the three items would occur (first and third scenario and/or second and third scenario):

- Failure of the fuel cladding. The two main causes of cladding degradation (TSPA-VA analyses) are mechanical failure and corrosion failure. The stainless steel clad fuel in breached waste packages and in defective waste packages will be available for dissolution, as will be a small percentage of the Zircaloy clad fuels. At about 100,000 years, waste packages that fail through both mechanical failure and corrosion will begin to expose more fuel for dissolution through continued degradation of Zircaloy cladding. At 1 million years, about 31 percent of the fuel will be available for dissolution in the base case cladding model.

- Degradation of the solid waste form. Following cladding failure, degradation of the commercial spent nuclear fuel occurs within a few thousand years, while degradation of the high-level radioactive glass form occurs over a period of 8,000 to 10,000 years (following failure of the stainless steel canister).

- Mobilization of radionuclides into aqueous solution, aqueous colloidal suspension, or gaseous form.

As discussed in Section 2.2 of Volume 4 and supported by the TSPA-VA analyses discussed in the previous list, the principal factors affecting the postclosure performance of the waste package include the following: water dripping onto the waste package, the humidity and temperature at the waste package, the chemistry of the water contacting the waste package, the integrity of the waste packages' carbon steel outer barrier, the integrity of the waste package's high-nickel alloy inner barrier, the integrity of spent nuclear fuel cladding, and the dissolution of the uranium oxide and glass waste forms. These principal factors impact the lifetime of the waste package and the release of radionuclides from breached packages. The waste package materials and waste form testing program is designed to reduce the uncertainty for the principal factors that affect the waste package performance and to study the containment provided by the waste package barriers and the spent nuclear fuel cladding.

The testing program provides data that supports the selection of barrier materials to prevent the release of radionuclides, supports the design of the engineered barrier system, and serves as input to the TSPA. The testing program assesses the behavior, short-term and long-term, of the materials comprising the waste package and the engineered barrier system and those that constitute the waste forms.

Testing the materials comprising the waste package supports design by enabling the selection and specification of barrier materials, and the specification of barrier thicknesses through the generation of material corrosion or degradation models. The waste package degradation models incorporate the important processes for corrosion of both the outer and inner barriers. The data generated from the materials tests serve as input for the analyses that support the adequacy of the reference design. These analyses are found in Sections 3.4.2 of

Volume 3. The analyses address corrosion modes for a variety of environments, information included from expert elicitation, juvenile failures, and long-term performance.

Testing the waste forms supports the waste package design by providing source terms for radionuclide release through the engineered barrier system to support safety analysis and performance assessment. Testing the commercial spent nuclear fuel and high-level radioactive waste glass forms has enabled the development of preliminary models for waste form degradation for use in the TSPA, discussed in Section 3.5 of Volume 3. Testing will continue on waste form materials to provide credible performance models.

Tests to date have shown that an outer barrier, corrosion allowance material of ASTM A 516, and an inner barrier, corrosion resistant material of Alloy 22, are the appropriate choices for the baseline design for the waste package and would meet the minimum containment requirements. Testing will continue on materials for both baseline and other design options (such as titanium) to allow trade-off, or comparison, studies for longer-life waste packages. The effects of design options under consideration are found in Section 4.5 of Volume 3.

5.1.4.1 Comprehensive Corrosion Tests on Waste Package Materials

Comprehensive corrosion tests are being conducted on specimens of waste package materials exposed to simulated repository environments. These tests will help select materials that resist corrosion under the expected conditions inside the repository. The tests are comprehensive, in the sense that many forms of corrosion can be tested: general corrosion, pitting, stress corrosion cracking, galvanic corrosion, and corrosion in crevices are just a few examples. In addition, parts of the long-term tests have counterparts in shorter-term tests for stress corrosion cracking and galvanic corrosion. The material presented here will discuss the following:

- Test Environment
- Test Specimens
- Complementary Short-term Tests
- Field Testing

Long-Term Test Environments and Inspections. The test environments are structured to simulate bounding conditions. The composition of water that might seep into the drifts or drip onto the waste packages is not known. However, in the absence of thermal effects, the water is expected to be similar to that extracted from Well J-13 because the potential repository and the well are in the same geologic unit. Formulas have been developed for setting up these test environments. The formulas were based on previous experience both at the Yucca Mountain Project and at Argonne National Laboratory where well water from Yucca Mountain was simulated using various salts. Since concentrated solutions are generally more corrosive than dilute ones, all of the bounding water environments are more concentrated than J-13 well water. The characteristics of the four bounding water environments have been estimated from the geochemical code and are summarized in the following list.

- The base case water, with a low concentration of ions (atoms or particles that have a positive or negative electric charge), will have a pH of 8.5 (a slightly basic solution) and contain 1,700 parts per million (ppm) of total dissolved solids. For the purpose of comparison, this water is estimated to contain 67 ppm of chloride ion.

- The base case water concentrated 100-fold, with a high concentration of ions, will have a pH of 10 and contain 146,000 ppm of total dissolved solids. This water is estimated to contain 6,700 ppm of chloride ion. This bounding environment represents what might happen with repeated plumes of rainwater descending toward the repository, encountering the host rock that has been heated by thermal radiation, evaporating, and leaving behind salts. Finally, after the temperature of the surface of the waste package cools to the boiling point of water, subsequent plumes of water could dissolve the salts and carry them to the surface of the waste package.

- The concentrated water was water treated with sulfuric acid to a target pH of around 2 (very acidic). This water is estimated to have a total dissolved solids content of 146,000 ppm and to contain 24,200 ppm of chloride ion. This bounding case represents the condition where certain species of microbes have produced acidic metabolic products. It also simulates the chemical changes that might take place in water that is confined in a crevice between two pieces of metal or between metal and a rock.

- The concentrated water was water with calcium hydroxide added to reach a target pH of 12. This water is estimated to have a total dissolved solids content of 132,000 ppm and to contain 20,900 ppm of chloride ion. This bounding case represents water conditioned by prolonged contact with materials containing cement, such as those used to line the drift wall or used in the invert material underneath the waste package.

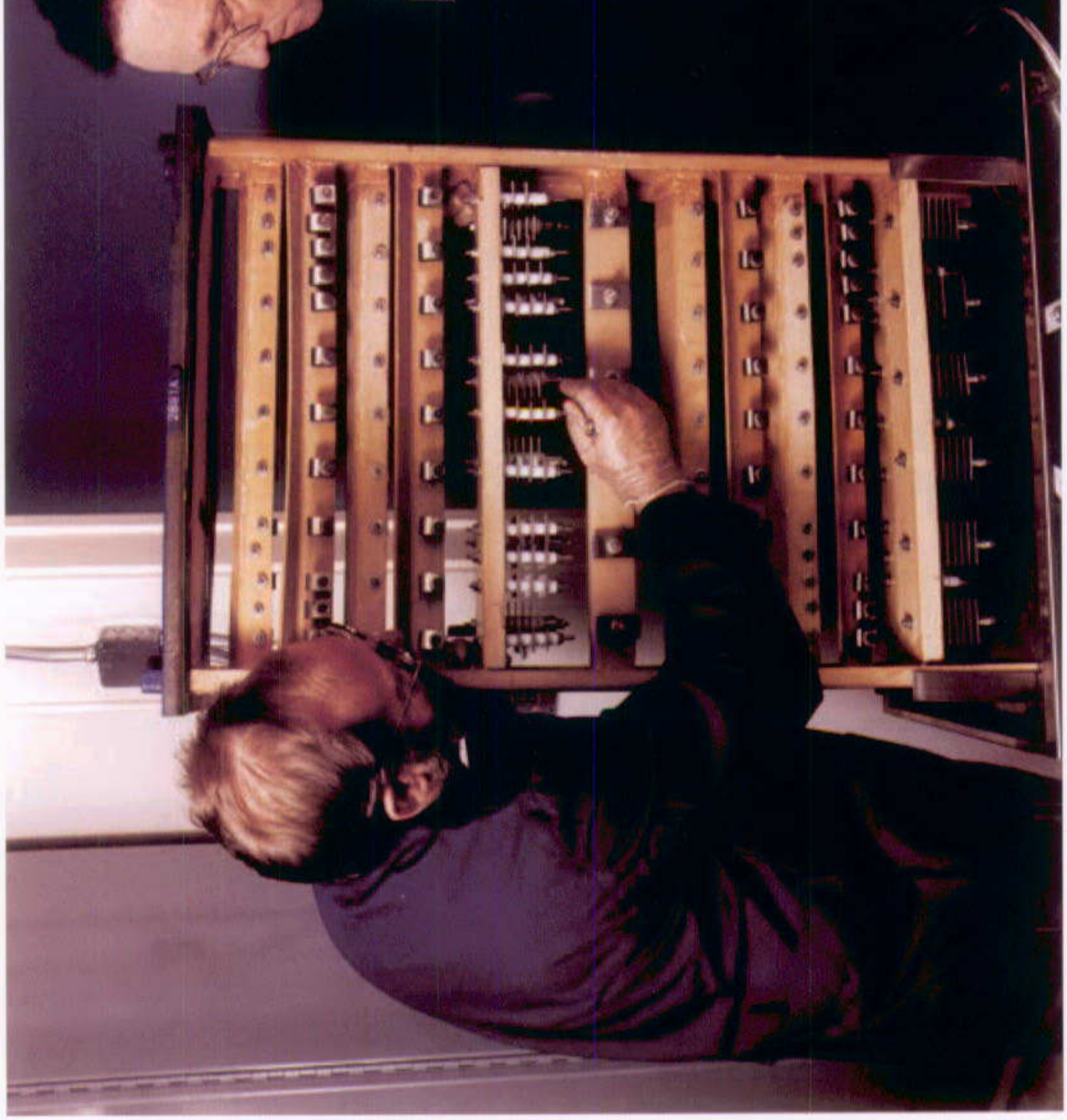
The four proposed bounding environments provide a range of pH (acid, neutral, and alkaline) and a range of ionic strength (dilute and concentrated). The ionic strength affects the conductivity of the water, which in turn affects local corrosion of the metal surface. From the point of view of metal corrosion, the chemical species normally found in the Well J-13 water and other groundwater near Yucca Mountain are expected to give the general reactions listed in the following bulleted list.

- **Anions (negatively charged ions)**
  - Chlorides and fluorides: usually aggressive, since they tend to break down protective passive films on many metals
  - Sulfates: somewhat aggressive to some metals; indifferent effects on others
  - Nitrates: combine with oxygen (oxidizing), therefore aggressive to some metals; promote the formation of a protective layer on other metals
  - Bicarbonates: tend to neutralize pH, indifferent effects or even protective effects if calcium and/or magnesium are present
  - Silicates: form a protective layer on metals

- **Cations (positively charged ions)**
  - Sodium, potassium, calcium, and magnesium ions-depends on which anions are also present
  - Heavy-metal ions-largely absent from groundwaters in the repository, but would have important effect on metal corrosion since they are oxidizing and promote the formation of an acidic environment. The effects of ions of iron produced from corroding materials (not directly associated with the waste package), such as bolts in the rock, steel mesh to support the drift, and conveyances to transport the waste packages, could be significant.

The corrosion response is a composite of the separate influences of these different ions. In many cases, it is the ratio of the concentration of aggressive ionic species to those of the protective or indifferent species that determine the overall corrosive properties of the water. Therefore it is important that the combined effects of all the various constituents of the water be present in the long-term corrosion test.

The specimens of the materials to be tested are attached to racks made of a non-reactive material such as Teflon, as shown in Figure 5-15. The racks are then placed in large tanks of water (about 3 ft square) that are also lined with a nonreactive coating. Each tank can hold six racks. The tanks have covers that fit well, but are not completely airtight. The tanks are filled to approximately 60 percent of their volume so that when the specimen racks are inserted, half of the specimens will be under the water, while half will be above the water and exposed to the air and steam. A few specimens will be at the water line. Because the tank is closed and reasonably tight, the air inside will become humid and be saturated with steam. Air from outside the tank will be passed into this vapor space to



FV205-14

Figure 5-15. Corrosion Test Rack

insure that the oxygen, carbon dioxide, and other gases, normally found in air, will be maintained in the steam vapor.

The *Engineered Materials Characteristic Report* (McCright 1998), was used in selecting two temperatures for the long-term tests, 60 and 90°C (140 and 194°F). These temperatures are bounding in the sense that the highest corrosion rates for the corrosion-allowance materials will be found in this range and the effects of localized corrosion and stress corrosion cracking will be greatest in this temperature range for the corrosion-resistant alloys.

The temperature and the level of the water are the only two parameters that are controlled in these tests. However, other parameters will be measured. These parameters include pH, chemistry of the water, oxygen content, and microbial activity.

At the scheduled time interval, a rack is withdrawn from the vessel so that the specimens can be inspected for corrosion. Starting in Fall of 1996, the following five planned intervals are planned for inspection: after 6 months of exposure, 1 year, 2 years, 3 years, and 5 years. (The sixth rack is for additional test specimens, if needed). At each withdrawal period, certain specimens will be weighed and the rate of corrosion calculated. Surfaces will also be inspected for evidence of localized corrosion. Some specimens contain welds, so that the weld and the area around the weld (heat-affected zone) will be carefully inspected. Depending on what is initially observed, appropriate types of analyses will be pursued to fully characterize the specimens. Samples of the water will be taken periodically to check for any changes in chemistry. These samples will also be characterized for the nature and extent of microbial activity.

Long-Term Test Specimens. Because the long-term test is also a comprehensive test for several modes of corrosion, the following three different specimen configurations are exposed in the test environment: one to examine general corrosion; a second, for corrosion in a crevice; and a third, for cracking and embrittlement. A "weight-loss" coupon, approximately 5 cm × 2.5 cm × 0.3 cm (2.0 in. × 1.0 in. × 0.125 in.) thick, yields information on the rate of general corrosion. When the specimen corrodes or oxidizes, a corrosion product like rust forms. This oxidation product is removed from the specimens before they are weighed. Further examination of the pattern of attack reveals whether the corrosion is uniform, or shows localized (small patches) attack such as pitting. A crevice specimen, which is slightly larger, is arranged so that a Teflon washer forms a tight crevice with the metallic specimen. Third, a specimen bent into the shape of a "U" is used to determine susceptibility to stress corrosion cracking (a phenomenon caused by stress and the environment) and embrittlement due to an increase in hydrogen content. The U-bend specimens are 0.16 cm thick and formed to be 3 cm high × 1.9 cm wide (0.062 in. × 1.25 in. × 0.75 in.). In all cases, the test specimens are carefully isolated from mounting bolts and hardware. The specimen designs and test procedures are based on specifications developed by the American Society of Testing and Materials including specifications G1, G30, and G46.

Some 13,000 specimens, in the three configurations, are being tested. The first materials to be tested are arranged in three categories as shown in the following bulleted list:

- **Corrosion-allowance materials**
  - Wrought carbon steel (ASTM A 516)
  - Cast carbon steel
  - Chromium-molybdenum alloy steel
- **Intermediate corrosion-resistant alloys**
  - Monel (nickel-copper alloy)
  - Copper-nickel alloy
- **Corrosion-resistant alloys**
  - Incoloy 825
  - Hastelloy G-3
  - Inconel 625
  - Hastelloy C-22, Inconel 622
  - Hastelloy C-4
  - Titanium Grade 12
  - Titanium Grade 16 (trace palladium)

For the first series of corrosion tests (referred to as Increment 1), the test specimens are kept within their respective categories to avoid any cross effects between corrosion products. In a later increment, cathodic specimens will be placed where the cross effects are desired. These tests will be discussed later.

For Increment 1, specimens of carbon and alloy steels were placed in tanks containing the dilute water and the concentrated water environments at the two test temperatures (thus requiring four tanks). The specimens of the nickel-based and titanium-based highly corrosion-resistant materials were placed in four more tanks similarly filled and heated. Then, specimens of these same nickel-base and titanium-base alloys, and specimens of the intermediate copper- and nickel-bearing alloys were placed in the acidified concentrate environment. Additional tests will be run on different combinations of materials and in different environments.

Tests for assessing cathodic corrosion have begun in the same environments as the single-metal specimens, but in separate vessels. The design of the cathodically coupled specimens simulates the way the waste package is fabricated (see Section 5.1.2.3).

Complementary Short-Term Tests. The long-term corrosion tests are a cornerstone for evaluating waste package materials. However, several other short-term activities interface with the long-term tests. For example, the short-term tests predict the relative susceptibilities of the candidate materials to localized corrosion, microbiologically influenced corrosion, stress corrosion, and cathodic corrosion. The long-term corrosion test evaluates the validity of these predictions for the longer term. The short-term tests are also important for modeling corrosion behavior. Microbiologically influenced corrosion is being evaluated by conducting tests of metal surfaces exposed to biological communities as a function of nutrient chemistry. Two other short-term tests are discussed in the following paragraphs: galvanic corrosion and stress corrosion cracking.

Cathodic corrosion is the enhanced deterioration of one of two dissimilar metals by contact with the other metal in an electrolytic solution, resulting in protection of the other metal. While a variety of degradation modes can occur in aqueous environments, cathodic corrosion of the outer barrier of the waste package is considered an important mode. Accelerated corrosion of the outer barrier may occur because of its contact with the more corrosion-resistant inner barrier while exposed to water in the crevice between the two barriers. The tests are concerned with evaluating the cathodic corrosion behavior of many different metallic couples in aqueous environments, by measuring the current between two dissimilar metallic materials and observing the corrosion that occurs.

Stress corrosion cracking is an environment-assisted phenomenon resulting from the combined interactions of tensile stress and a corrosive environment. Environments causing stress corrosion cracking are usually aqueous, and can be either condensed vapor or solutions. Hydrogen embrittlement is also a form of environment-induced failure that results most often from the combined action of hydrogen and residual or applied tensile stress. While several mechanisms of the two phenomena have been proposed based on numerous studies, no single unique mechanism has been widely accepted. Therefore, the proposed test program is focused on evaluating the stress corrosion behavior of susceptible waste package materials under measured and/or controlled electrical currents in simulated repository environments. The resultant data will help the understanding of the cracking process in materials of interest. This information will be used to develop and validate the models for stress corrosion cracking for the long-term performance of the waste package materials.

Field Tests and Natural-Analog Assessment. Field tests will characterize how candidate materials for waste packages degrade after they have been exposed to field environments at Yucca Mountain. These studies are primarily being performed to learn how the host rock of the repository responds to being heated. Specimens of the waste package materials are used in these studies to determine the long-term degradation of the materials under the elevated temperature conditions at Yucca Mountain. These specimens see the environmental change, particularly temperatures, over time compared to the laboratory experiments that are run at fixed and constant environmental conditions.

In each study, test specimens are placed in regions of the rock that are well characterized with respect to temperature and relative humidity. Test specimens are attached to the packets that are set in boreholes for all of the field studies. Temperature and relative humidity are measured near the attached specimens.

Natural analogs are evaluated to support the prediction of the long-term behaviors of the barrier materials. Natural analogs are phenomena that occur naturally—sometimes over extremely long periods of time—and cannot easily be duplicated in a laboratory.

The data on corrosion generated from the field tests and the natural analog assessments are used in activities related to the performance assessment, materials selection, model development, and repository design. This work described in this section addresses aspects of the subissue "What are the effects of corrosion on the lifetime of the containers and the release of radionuclides to the near-field environment?" This subissue is associated with the Container Life and Source Term Key Technical Issue (NRC 1998a). Status information about this subissue is summarized in Volume 4, Section 4.3.3.4.

5.1.4.2 Waste Form Testing Program

Waste packages are being designed to contain radioactive waste for thousands of years. However, the packages can potentially degrade as they react to the repository environment over time. As the waste packages degrade, the waste forms inside could be exposed to the conditions inside the repository. The behavior of the waste forms when exposed directly to the repository environment determines the amount and rate of release of radioactive materials; therefore, the behavior of the waste forms must be evaluated under simulated repository conditions. Solutions used for testing are based on simulated Well J-13 water appropriately modified to meet specific test objectives. This work described in this section relates to the resistance of spent nuclear fuel and high-level radioactive waste glass and addresses aspects of two subissues concerning resistance to degradation associated with the Container Life and Source Term Key Technical Issue (NRC 1998a). Status information about this subissue is summarized in Volume 4, Section 4.3.3.4.

This section discusses the testing done for fuel coming from commercial and DOE nuclear reactors and other high-level radioactive waste that has been encased in borosilicate glass or ceramics. The work described in this section relates to the resistance of spent nuclear fuel and high-level radioactive waste glass and addresses aspects of two subissues concerning resistance to degradation associated with the Container Life and Source Term Key Technical Issue (NRC 1998a). Status information about this subissue is summarized in Volume 4, Section 4.3.3.4.

Commercial Spent Fuel Testing. Two types of commercial spent nuclear fuel tests are performed: dissolution and oxidation. Dissolution tests measure the rate at which radionuclides are released when the spent nuclear fuel is directly exposed to water. Data obtained from the dissolution tests include:

- Dripping rate of water
- Integrated flow volume
- Estimated surface area of the sample being exposed to the water
- Estimated time that the water stays in contact with the surface of the spent nuclear fuel
- Chemical and radionuclide states of the aqueous solution

As the volume of aqueous solution accumulates from the ongoing tests, the materials that are suspended, not dissolved, in the solution will be characterized in detail.

Oxidation tests are designed to determine the oxidation characteristics of spent nuclear fuel at the temperatures anticipated after the waste package has degraded and after the cladding has degraded or failed. These tests also identify the influence of important fuel characteristics (e.g., fission gas release, burnup, and fuel type) and atmospheric variables (e.g., moisture content and radiation field) on oxidation rates and mechanisms.

DOE-Owned Spent Nuclear Fuel Testing. DOE-owned spent nuclear fuel testing is being performed by the National Nuclear Fuel Program under a materials science program designed to study the release rates, materials analysis, and localized corrosion in order to determine the response of DOE-owned spent nuclear fuel to groundwater exposure in order to provide specific data on what fissile products are released and the release rate. These tests will focus on fuel characteristics that may cause release rates to differ from those of commercial spent nuclear fuels within the repository.

High-Level Radioactive Waste Glass Testing. High-level radioactive waste glass currently being tested includes glasses from Savannah River Technology Center and the West Valley Demonstration Program. Both doped and real radioactive glasses are being tested. Glass from Handford will be tested when it becomes available. The best available methods to predict long-term performance of this waste form are as follows:

- Testing at elevated temperatures, which accelerates reaction rates

- Testing at high surface area-to-volume ratios, which accelerate rates proportionally to the exposed surface area of the glass

Flow-through tests and closed-system tests are planned. Additional tests may be performed that are not yet described, based on data collected as the work progresses.

Two types of flow-through tests will be performed. The first measures the rate at which the glass dissolves; the second test observes how other materials in the system affect this rate. In the first set of experiments, the glass is immersed in a fluid that has a constant pH and continuously flows past, and reacts with, the sample. The flow rate is kept fast enough so that the amount of glass dissolved will not cause the material to come out of solution and cause secondary phase to form. If the solution were to become supersaturated, the concentrations of elements in the solution downstream from the glass would not represent the total amount of elements released from the glass. Then the results would have to be corrected for the amount of a given element that has precipitated in secondary phases. Earlier studies show that the secondary phases are small and difficult to identify; therefore, only estimates of the amounts of materials for these phases may be possible.

The tests will be performed on a representative set of glass compositions, based on estimated compositions of actual waste glasses furnished by the Savannah River Technology Center. The dissolution of waste glasses is sensitive to the oxidation state of the elements they contain, because many of the elements such as iron, manganese, and uranium have multiple valence states. In addition, it is almost impossible to control oxidation state at low temperatures. As a result, testing materials such as basaltic glass may better indicate the effect of glass composition on the dissolution rate constant. The tests will be performed over a range of temperatures and pH values that bound anticipated repository conditions.

A second type of flow-through test will investigate the effects of other repository materials, such as metals contained in the waste package, on the corrosion behavior of glass. In these tests, metals such as iron will be added to the solutions, either as soluble salts or as corrosion products such as iron oxides.

These closed-system tests will provide further quantitative information on the solution composition and the rate of dissolution. In the closed-system tests, as the glass dissolves, glass constituents build up in solution. This buildup slows the rate of reaction. Unlike the flow-through tests where the solution leaves the glass, in the closed system tests the fluid remains in contact with the glass and the concentration of material in solution builds up until the solution becomes supersaturated. At this point, secondary phases form.

Analyzing the solution over time and assessing the secondary phases that precipitate will be used to validate the rate constants determined in the flow-through tests.

5.1.4.3 Material Performance

Evaluating material performance consists of several activities. These activities include reviewing and analyzing data from test programs and searching for and analyzing published data. Based on these analyses, models are developed for use in the performance assessment calculations. The waste package materials and the waste forms themselves are tested and modeled for performance. This section briefly describes each of these activities.

Degradation-mode surveys and information bases represent a consolidation of information related to the expected performance of materials used in fabricating the waste packages. Degradation-mode surveys compile information on failure modes of materials, notably forms of corrosion. These surveys also document oxidation and embrittlement phenomena and evaluate this information as it pertains to Yucca Mountain. Information bases are collections of information on characteristics of candidate materials, such as physical and mechanical properties, that are important to the waste package design.

Testing materials for their resistance to corrosion and evaluating their physical properties make up the largest effort in the material-performance area. The multiple-barrier design of the waste package and the number of options for designing the repository require a comprehensive testing program to evaluate how materials perform under the wide range of conditions anticipated in the repository. Several candidate materials are being considered for the containment barriers. For example, the waste package has an outer barrier made of a corrosion-allowance material and an inner barrier made of a corrosion-resistant material. The testing program thoroughly evaluates each material.

The performance of the various materials must be modeled. The modeling effort serves the following two major purposes:

- Supports the recommendations for selecting certain materials. A key selection criterion is the predictability of the performance of the material.

- Furnishes the information about how the selected material will perform in the repository environment. Consistent with ASTM C-1174-91, repository-relevant data from the testing activities is interpreted with an understanding how materials behave.

Examples of models developed for use in the TSPA include the following:

- General and localized corrosion of materials used for the containment barriers

- Degradation of waste package components, such as inner basket materials and neutron absorbers

- Solubility of spent nuclear fuel and radionuclides

- Oxidation of spent nuclear fuel

- Degradation of high-level radioactive glass

The details of the models have been provided in the *Engineered Materials Characteristic Report* (McCright 1998). Model predictions are compared to performance information in the literature and from that the need for, and type of, long-term tests are projected. Where applicable, natural analogs, will be used to partially validate long-term predictions.

5.2 UNDERGROUND PORTION OF THE ENGINEERED BARRIER SYSTEM

The underground portion of the engineered barrier system comprises all elements that are outside the waste packages. These elements include the following:

- The steel supports that hold the waste packages
- The piers upon which the steel supports rest
- Certain aspects of the drift invert that supports the pier
- The way the emplacement drifts are arranged and excavated

Figure 5-16 shows the locations of these elements within an emplacement drift.

This section describes the emplacement drifts, the inverts that form the floor of the drift, the piers, and the steel supports that support the waste package. Several other components to enhance the performance of the system are being considered. These options are discussed in Section 5.3.

5.2.1 Emplacement Drifts

10 CFR 60.133(a)(1) requires that, "the orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are a part of the underground facility shall contribute to the containment and isolation of radionuclides." This part of the discussion describes how the design of the drifts meets these requirements.

The structure of the host rock influences how the emplacement drifts will be situated. To the extent practical, drifts will be oriented so they are not parallel to long, continuous fractures or cracks that naturally occur in the host rock. This practice should promote drift stability. However, orientation of drifts according to structure orientation is not straightforward. The experience gained from excavating the Exploratory Studies Facility indicates that the orientation of the fractures is not constant throughout the primary emplacement area. Because the drifts should be parallel and not at different orientations, the final orientation of the drifts will likely be a compromise, or "best fit" arrangement.

The openings of the emplacement drifts are part of the engineered barrier system to the extent that their physical arrangement can affect the performance of the repository. The excavated openings are circular, which is the most inherently stable configuration for an underground opening. Shapes with corners or other irregularities are prone to higher stress concentrations and are generally considered less naturally stable than circular shapes. Another favorable feature of a round opening is that, if liquid water seeps into a round opening, it is more likely to stay in contact with the drift surface and run around the perimeter of the drift than to form a droplet and drip onto a waste package. As discussed in Volume 3, water dripping on the waste package is one of the most important causes of eventual waste package failure.

The layout of the subsurface has also been developed to promote containment of the radioactive waste. The drifts are widely spaced, which reduces the extraction ratio. The extraction ratio is the excavated area divided by the total area and is a measure of how much rock has been removed. Higher extraction ratios place more stress on the remaining rock to support the overlying rock mass. Conversely, limiting the extraction ratio creates lower stresses, and therefore allows a more stable drift configuration. The extraction ratio is calculated as follows:

(5.5-m drift diameter per 28-m drift spacing) $\times$ 100 = 19.6 percent

This low extraction ratio of 19.6 percent, combined with the relatively high strength of the rock mass, will create stable conditions and limit the possibility of collapse of the overlying rock. Such a collapse could impair the repository's ability to contain the waste because the rock mass above the repository would be disturbed and become more permeable, allowing water to move toward the emplaced waste.

In addition to having a low overall extraction ratio, the repository system is arranged so that water moving downward through the repository zone and entering a repository opening will not be preferentially directed toward emplaced wastes. The repository layout will be configured so that:

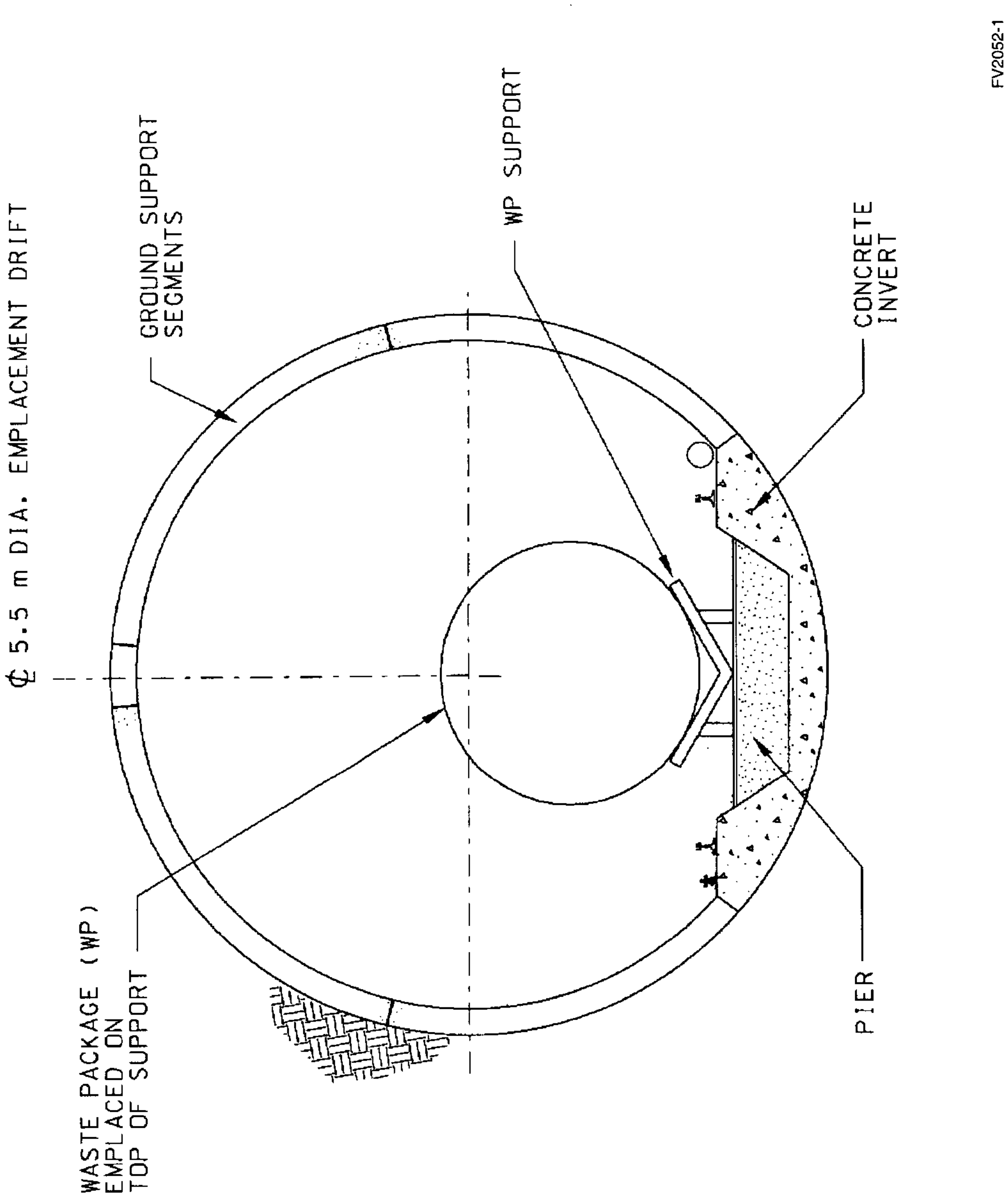


Figure 5-16. Emplacement Drift Section at Waste Package Support Location

- Water that enters the emplacement drifts can drain, by gravity, out of the emplacement drifts into the mains and, once out of an emplacement drift, cannot reenter, by gravity flow, another emplacement drift.

- Drifts excavated above the emplacement level will not be directly connected to an emplacement drift to prevent water that enters the overlying drift from flowing through a man-made opening into the underlying emplacement drifts.

- Drifts excavated above the emplacement level will be sloped so that water that enters the drift can flow away from the emplacement area.

The depth of the underground facility has been selected to comply with siting requirement 3.7.2.2 F of the *Mined Geologic Disposal System Requirements Document* (DOE 1997c, DOE/RW-0404P April 1996). This requirement states that emplacement areas have at least 200 m (656 ft) of cover. Using the minimum depth maximizes the vertical distance from the emplacement area to the underlying water table. Because the eventual path of radionuclides is downward from the repository level to the water table, maximizing this distance should also increase the travel time for radionuclides.

10 CFR 60.133(f) states the following:

> The design of the underground facility shall incorporate excavation methods that will limit the potential for creating a preferential pathway for groundwater to contact the waste packages or radionuclide migration to the accessible environment.

Using tunnel boring machine technology to develop emplacement drifts is the best available means to comply with this requirement. Any excavation method that makes an opening will change the surrounding rock. However, excavating with the tunnel boring machine and fully lining the drift immediately after excavation should result in minimal fracturing of the rock units, and thus comply with this requirement.

The arrangement of the subsurface drifting is discussed further in Section 4.2.

5.2.2 Drift Invert

The information in this section addresses aspects of the NRC Key Technical Issues of *Repository Design and Thermal-Mechanical Effects* (NRC 1997c), and *Radionuclide Transport* (Sagar 1997). The drift invert is the bottom of the emplacement drift. This term also applies to the portion of the ground control system that rests on the bottom of the drift. The invert is a reinforced precast concrete unit that supports the waste package; associated structures keep the waste package from resting directly on the invert. Figure 5-16 shows the location and shape of the invert.

The invert, while a part of the ground control system, is also a part of the engineered barrier system because its features have some effect on the retention of radionuclides in the emplacement drift. This performance feature could be enhanced. For example, the invert could be fabricated using materials that "sorb," or preferentially hold onto, certain radionuclides. Such a measure could be used to slow down the ultimate release rate of those radionuclides that are susceptible to sorption. The reference design does not incorporate such "sorbers." Their use may be considered in the evaluation of options for use in the LA design.

The drift invert, in the reference design, is a structural member that forms part of the full ring of concrete that makes up the drift lining. It also serves as the roadbed for both construction traffic during excavation and for the emplacement gantry during the emplacement operation. The invert segment design has included structural evaluations in the *Emplacement Drift Invert Structural Design Analysis* which incorporate combined thermal, in-situ, and seismically induced loads to show that the invert will remain intact and functional (CRWMS M&O 1998d).

As an alternate to the concrete invert design, an all-steel invert system has also been carried in the design. The steel concept was carried forward to mitigate the effects on the design of a possible elimination or limitation on the use of concrete in the emplacement environment. Concrete is being evaluated for its effects on postclosure performance, and its use may be limited or eliminated. An all-steel ground support alternative, incorporating the steel invert, has been carried in the VA reference design to limit the impact on design of such a decision (CRWMS M&O 1998d).

The arrangement of the subsurface drifting is discussed further in Section 4.2.

5.2.3 Support Structure for the Waste Package

The support assembly for the waste package is a pier that serves as a foundation on which to set the support that holds the waste package. Together, the pier and support hold the waste package off the invert, which serves two important purposes: preventing water from contacting the waste package and helping cool the waste package by allowing air to circulate around it. The requirements to prevent water from contacting the waste package are stated in the *Engineered Barrier Design Requirements Document*, Parts 3.2.3.3.A.8.b and 3.2.3.3.A.8.c (YMP 1994a).

Preliminary designs have been developed for the support structures. These designs are intended for use with the preliminary designs developed for the drift lining. The lining is composed of six precast concrete segments that together form a full-circle lining in the drift. The invert segment is a part of the ground support system and also serves to support the waste package support system. An alternate preliminary design using all steel and no concrete has also been developed. Although a pier has not been developed to work with this alternate design, a concept for an all-steel pier has been developed. The designs for the pier and waste package support have been analyzed for structural stability and thermal performance in the repository environment.

A modular design has been developed for the waste package support assembly. The decision to use a modular design responds to requirements 3.2.5.2.1, 3.2.5.2.4.C, and 3.2.5.2.8.A.2 in the *Engineered Barrier Design Requirements Document* (YMP 1994a). A modular design allows flexibility in arranging the waste packages in the drifts and in replacing individual components of the support system should they be damaged in a waste package handling accident. If such an accident happens, it is less critical for the support structure to be damaged than for the waste package to be breached. Therefore, the support structure is designed to yield should a waste package handling accident occur.

The waste package support assembly has the following two parts: a pier and a support. The pier, as shown in Figure 5-17, is fabricated from seven steel plates, a half section of steel pipe, two steel bars, two hooks, and concrete. The plates, section of pipe, and steel bars are welded together to form a shell into which the concrete is poured. Steel pins, placed in the holes in the top steel plate, allow holes to be formed in the concrete. The steel pins are removed once the concrete begins to set. The base of the support will be inserted in these holes.

Each component of the pier serves a purpose as follows:

- Holes on the surface are used to position and attach the support
- Bottom steel bars help position the pier on the invert
- Shell, made of the steel plates, protects the concrete from being chipped during handling and allows the pier to be lifted

Each waste package support is fabricated from two pieces of rectangular steel tubing, two sections of steel pipe, and two circular steel bars, all made of carbon steel. At least two such supports or cradles will be used to hold each waste package. The sections of rectangular steel tubing are each angled on one end to create a V-shape when put together. This tubing also has small holes for drainage and larger holes that are used for lifting. The sides of the support are made of steel pipe. The pieces are welded together to form the support.

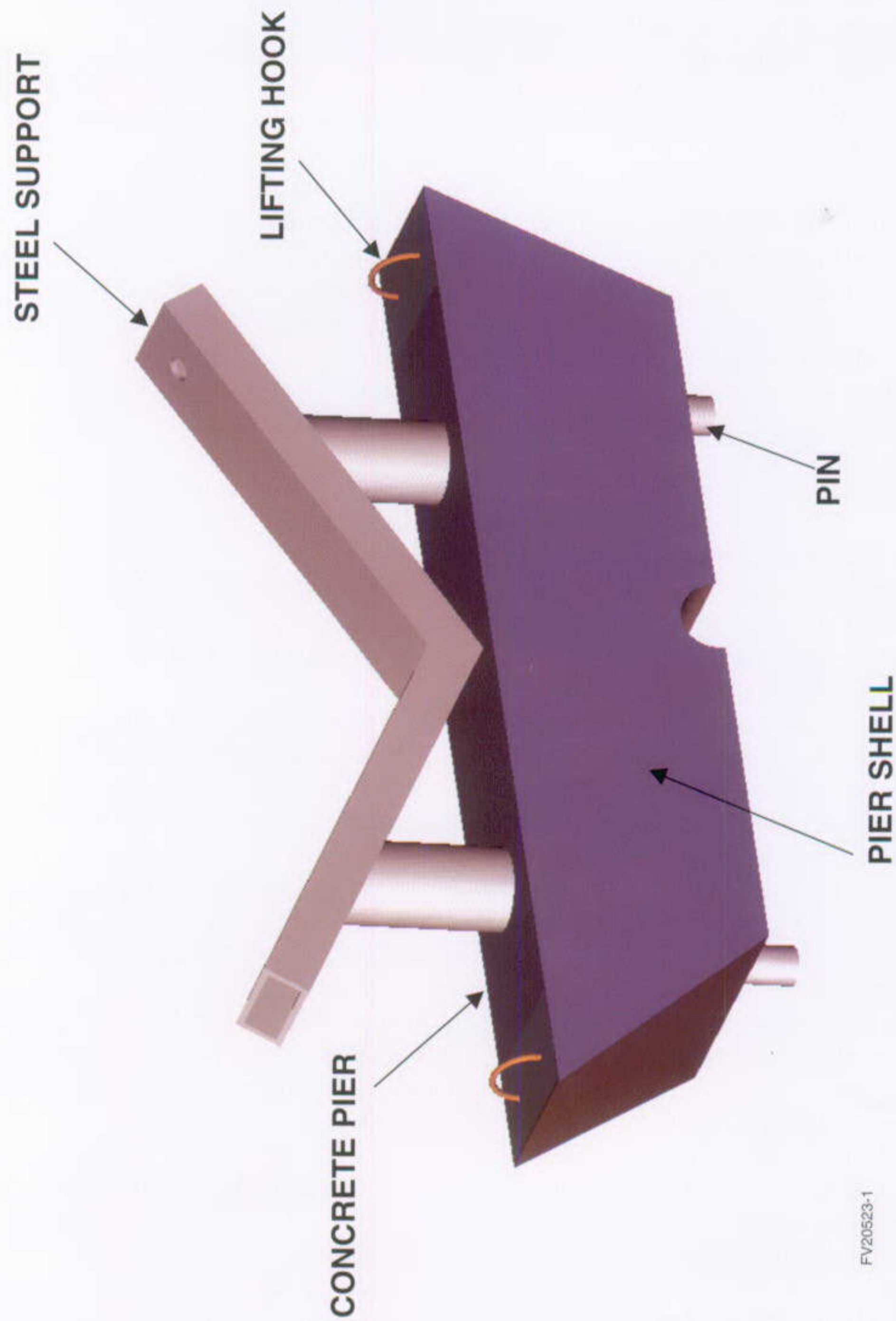


Figure 5-17. Pier and Support Assembly

The rectangular steel tubing serves as the saddle for supporting the waste package. The V shape will accommodate any diameter waste package. Rectangular tubing is used for its ability to distribute the weight of heavy loads. Carbon steel construction provides a relatively soft interface resulting in a larger contact area between the support and the waste package allowing greater distribution of loads.

As indicated in the *Controlled Design Assumptions Document*, Key 016, the waste package support and pier are designed to remain intact at least through the period of retrievability, up to 100 years after the start of waste emplacement (CRWMS M&O 1998b). General corrosion of the steel during this relatively short period will be minimal because the relatively high drift temperatures preclude the presence of liquid water. Therefore, the support and pier should easily meet this goal.

The support assembly for the waste package has the following two major interfaces: one between the support and outer surface of the waste package; the other between the pier and the concrete invert. The waste package support reduces stress on the outer surface of the waste package because it has been designed to deform under impact of a dropped package and has been fabricated out of a similar material to minimize material interactions. In *Repository Design Emplacement Drift Ground support Pre-Case Concrete Lining*, the center-to-center spacing of the supports is assumed to not exceed 1.5 m (4.5 ft) (CRWMS M&O 1997x), ensuring that every waste package would rest on at least two waste package supports when placed anywhere along the emplacement drift.

The second interface is between the pier and the precast concrete invert. The invert interfaces with the bottom plates, side plates, and bottom bars of the pier. As discussed before, the bottom bars are used to position the pier. The steel plates, side and bottom, provide a flat, somewhat ductile, surface between the concrete of the pier and the concrete invert. This surface is important for preventing rocking or tensile stresses in the concrete that may develop if the concrete surfaces are uneven. The dimensions of the pier have been set to match the dimensions of the precast concrete invert to ensure a good fit between components.

5.3 ENGINEERED BARRIER SYSTEM DESIGN OPTIONS

Several engineered barrier system enhancements are being evaluated. Current evaluation of options may prove prudent should a performance standard be established and subsequent overall performance assessments indicate that engineered enhancements are needed to meet the performance standard or are desirable from a safety margin perspective. Following publication of a performance standard, options currently under evaluation will be incorporated into the LA design or eliminated from further consideration, as appropriate, before any license application is submitted.

The features being evaluated for potential performance impacts are as follows (Figure 5-18):

- Emplacement drift backfill
- Drip shield, with backfill
- Ceramic coating of the disposal container (waste package), with backfill

Each measure is discussed in the following sections. Section 4.5 of Volume 3 presents the performance assessment associated with each of the engineered barrier system enhancements under consideration, as well as a performance assessment for combined enhancements. Costs of these options are discussed in Appendix G of Volume 5. The work remaining to be completed prior to submittal of any license application is discussed in Section 3.3 of Volume 4.

5.3.1 Emplacement Drift Backfill

Emplacement drift backfill is an option as well as a component of the other options. Backfill in the emplacement drifts could have at least the following three beneficial impacts on performance:

- Provides a measure of mechanical protection for the waste packages. Though no general failure of the rock mass in the emplacement

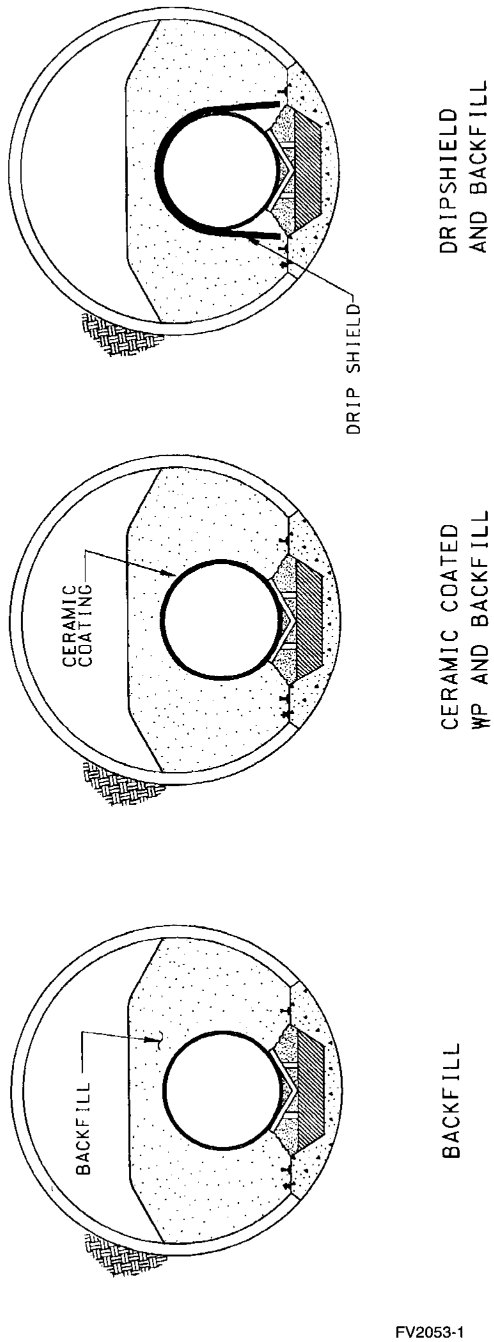


Figure 5-18. Engineered Barrier System Options for the Viability Assessment

horizon is anticipated, the concrete lining of the drifts will eventually deteriorate, allowing the rock in the crown of the emplacement drifts to fall into the drift opening. The resulting impact of falling rock on the waste packages may cause a percentage of the containers to breach earlier than if they were not impacted. This is especially true if the rockfalls do not happen until after the initially robust waste packages are made more vulnerable by extensive corrosion.

- Lowers the relative humidity conditions at the surface of the waste packages. Emplaced backfill will form an insulating layer around the waste packages. Because the waste in the containers will continue to generate heat for thousands of years, the insulating effect will raise the temperature at the waste package surface. Higher temperatures should result in lower relative humidity at the waste package surface. If the humidity at the waste package surface remains below the threshold for humid air corrosion, the onset of corrosion will be delayed.

- Prevents dripping water from directly contacting the waste packages. After the thermal pulse of the waste has passed, the drift environment will return to a below-boiling condition when liquid water, if present, could drip directly onto the surface of the waste package. The water will either boil away, if the container surface is still hot enough, or evaporate leaving behind entrained minerals. The mineral depositions on the container surfaces may promote corrosion. Backfill above the surface level of the waste package will preclude the dripping water from directly contacting the container surface.

The use of backfill has not been made a part of the reference design because it is a high cost operation and because there has not been a compelling case made that its performance benefits outweigh the costs. In addition, the large variability in the heat characteristics of the waste stream requires that, if used, backfill not be placed until very late in the life of the facility, just prior to closure. This is due to the insulating effect that backfill has on the waste packages. If backfill were placed too soon after waste emplacement, the temperature of at least a portion of the spent fuel cladding would exceed the thermal goal of 350°C (662°F). Exceeding this temperature may adversely impact the integrity of the Zircaloy cladding of the fuel rods and reduce its effectiveness as a potential engineered barrier system element. Cladding as an engineered barrier system element is discussed in Section 5.5.2 of Volume 3.

A number of questions regarding backfill still need to be answered. The ability of the backfill to protect the waste package and other components against rockfall needs to be tested to ensure that the design would work. Further, although there is considerable information on the properties of backfill under the average flow conditions, the response to the full range of possible conditions that might result from focused or episodic flow has not yet been fully determined. With regard to feasibility, backfilling of mines is a part of standard practice. What needs to be demonstrated is that emplacement to the specifications needed for repository functions can be accomplished. Consequently, rather than high confidence in the representation of the backfill, current confidence is moderate.

If required, backfill would be emplaced by remotely operated equipment due to the radiation levels in the emplacement drift environment. A conceptual design for this equipment has been developed. The as-placed characteristics of the backfill are limited by the placement method. No compaction is feasible within the current design concept. The backfill material would be granular, dry, and free-flowing and could be handled by standard industry bulk-material-handling equipment. Either quartz sand or crushed, screened, mined rock from the repository excavation would be used as the backfill medium.

The placement process in an emplacement drift would begin with cooling the drift, using air flow provided by the subsurface ventilation system. When the temperature of the air exiting the drift is at or below 50°C (122°F) the backfill material would be hauled from the surface by conventional rail and loaded into a hopper near the drift entrance. The fill would be moved to the placement machine, a simple conveyor device, using a self-unloading shuttle car, loaded at the drift entrance, and moved into the drift to the rear of the placement machine (see Figure 5-19). As the level of the fill reaches the point where the containers are covered by at least 0.6 m (about 2 ft) of fill, the placement machine pulls back slightly. This operation starts at the end of the drift farthest from the entrance and retreats toward the entrance as work progresses. The operation is continuous while the placement machine has fill material available. When the backfill runs out, another load is brought by shuttle car. The operation is remotely controlled by personnel who monitor the operation by closed-circuit television.

There is no need to completely fill the drift. The open area at the top of the fill provides a pathway for ventilation air to continue to flow through the drift. This will keep the dust of the operation from obscuring the view of the operator and keep the drift temperature low enough to protect the equipment. The exhaust raise is protected from being covered by the backfilling operation.

Additional design scenarios incorporating the use of backfill are discussed in Section 8, Major Design Alternatives, in this volume.

5.3.2 Drip Shields

The information in this section addresses aspects of the NRC Key Technical Issue of Container Life and Source Term (NRC 1998a). A drip shield is an optional engineered barrier system component that would be positioned above the waste package. The purpose of the drip shield is to prevent water from dripping directly onto the surface of a waste package. To be an effective part of the engineered barrier system, the drip shield must be long-lived and remain intact throughout the thermal pulse period to be available to divert water when below-boiling conditions return to the emplacement drifts. Consequently, construction issues such as materials, configurations, as well as the methods and equipment needed to place the drip shields, are being evaluated.

Placing the drip shield over the waste package at the time of emplacement is preferred. If drip shield placement were deferred until closure, the drip shield placement machinery would have to pass beyond emplaced waste packages to access the containers farthest from the entrance. While this is possible, it is not desirable because the recovery of the placement unit could be problematic if it breaks down far into the drift. Placement of the drip shield at the time of emplacement would require that drip shield expenditures be made much earlier in the repository's life cycle than if their placement were deferred until closure. Any thermal impacts, such as retardation of heat rejection by the waste package to the surrounding drift walls, will be more pronounced if the drip shields are placed concurrently with waste package emplacement.

The drip shield option brings with it the requirement to backfill the emplacement drift because of the longevity required of the drip shield. The backfill will provide mechanical protection to the drip shield as the emplacement drift lining ultimately fails and the rock in the crown falls into the drift. The placement of backfill is described in Section 5.3.1.

Constructability Analysis of Backfill and Drip Shield Configurations is a design analysis on engineered barrier system options for the VA that considered two different drip shield configurations (CRWMS M&O 1998a). The objective of the analysis was to assess the ability to safely and cost-effectively construct various engineered barrier system options.

One configuration involves placement of a drip shield directly on top of the waste package. Therefore, the weight of the drip shield rests on the package and, in turn, is supported by the waste package supports. In this scenario, the drip shields could be placed on the waste packages at the time of waste package emplacement or at a later time. Backfill would be placed over the drip shield/waste package arrangement just prior to closure. Placement of this drip shield option would be similar to the placement of the waste packages, and the emplacement equipment would likewise be similar.

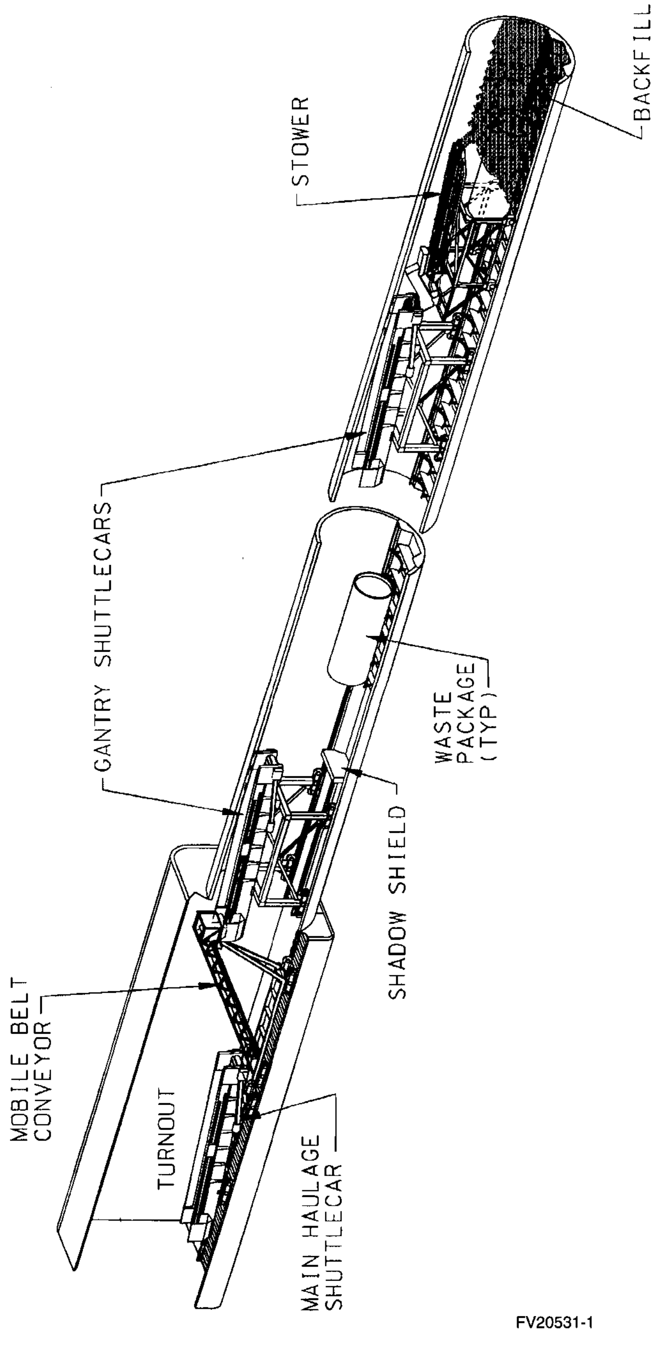


Figure 5-19. Emplacement Drift Elevation—Backfill Operations

The second drip shield option involves sequential placement of emplacement drift backfill with placement of drip shield segments that lay on top of the backfill. This method could be carried out only just prior to closure. The drip shield segments have ends that "interlock" to form a continuous barrier over the backfill. Though technically possible to install, this option would be much more difficult to implement than the above-described configuration.

The drip shields in both configurations are fabricated of Alloy 22. Alloy 22 is an extremely corrosion-resistant metal that should have a very long life in the expected drift environment. The drip shields would be 2 centimeters in thickness. Figures 5-20 and 5-21 show the two drip shield concepts described in this section.

There are several uncertainties regarding this barrier system. First, its effectiveness in diverting flow depends upon the longevity of the drip shield and of the backfill that complements it. Current confidence in the long-term stability of candidate drip shield materials (ceramics and corrosion-resistant metals) is moderate. Thermodynamic information on fired clay and alumina ceramics indicates that these will last millions of years; however, thermodynamic information for the particular ceramics under consideration by DOE has not yet been compiled and the bounding values of the time constants are not yet established.

Secondly, the flow properties of the drip shield/backfill combination need to be demonstrated for the full range of flow conditions. There is moderate to high confidence that this barrier system will prevent both advective and diffusive flow from reaching the waste package, but the specific response of this system to high fluxes needs to be determined.

Finally, because the specifications for the design, fabrication, and emplacement of this system have not yet been defined, current confidence in this system is moderate at this time. Uncertainties surrounding the ability and ease of construction have yet to be demonstrated before relying on this feature as a practical application.

Additional design scenarios that incorporate the use of drip shields are discussed in the major design alternatives, Section 8 of this volume.

5.3.3 Ceramic Coatings for the Waste Packages

The information in this section addresses aspects of the NRC Key Technical Issue, Container Life and Source Term (NRC 1998a). As part of the engineered barrier system, the waste package is being designed to work with the natural system in containing the radioactive waste. As a result, features that can extend the life of the waste packages are being considered. Ceramic coatings potentially could make the waste packages last longer by slowing the rate at which the packages will corrode. Work has begun at Lawrence Livermore National Laboratory to evaluate ceramic coatings as a design option and to determine the types of coatings that can be fabricated commercially.

The following three major topics are presented in this section:

- Ceramic materials and their application
- Experiments to test specimens under controlled conditions
- Potential limiting factors for using ceramic coatings

5.3.3.1 Ceramic Materials and Their Application

Ceramic oxides are promising as additional corrosion-resistant materials for the waste packages. Many available ceramics would be stable under the environmental conditions predicted for the repository. The candidate ceramic oxides have many of the following advantages: they are very stable; they are fully oxidized and therefore will not degrade due to oxidation; they are unaffected by salts; they do not conduct electricity and so are unaffected by galvanic corrosion; and, because they are inert, they are not a source of food for microbes. The materials currently being tested are magnesium aluminate spinel, aluminum oxide, and titanium dioxide, with spinel being the major focus.

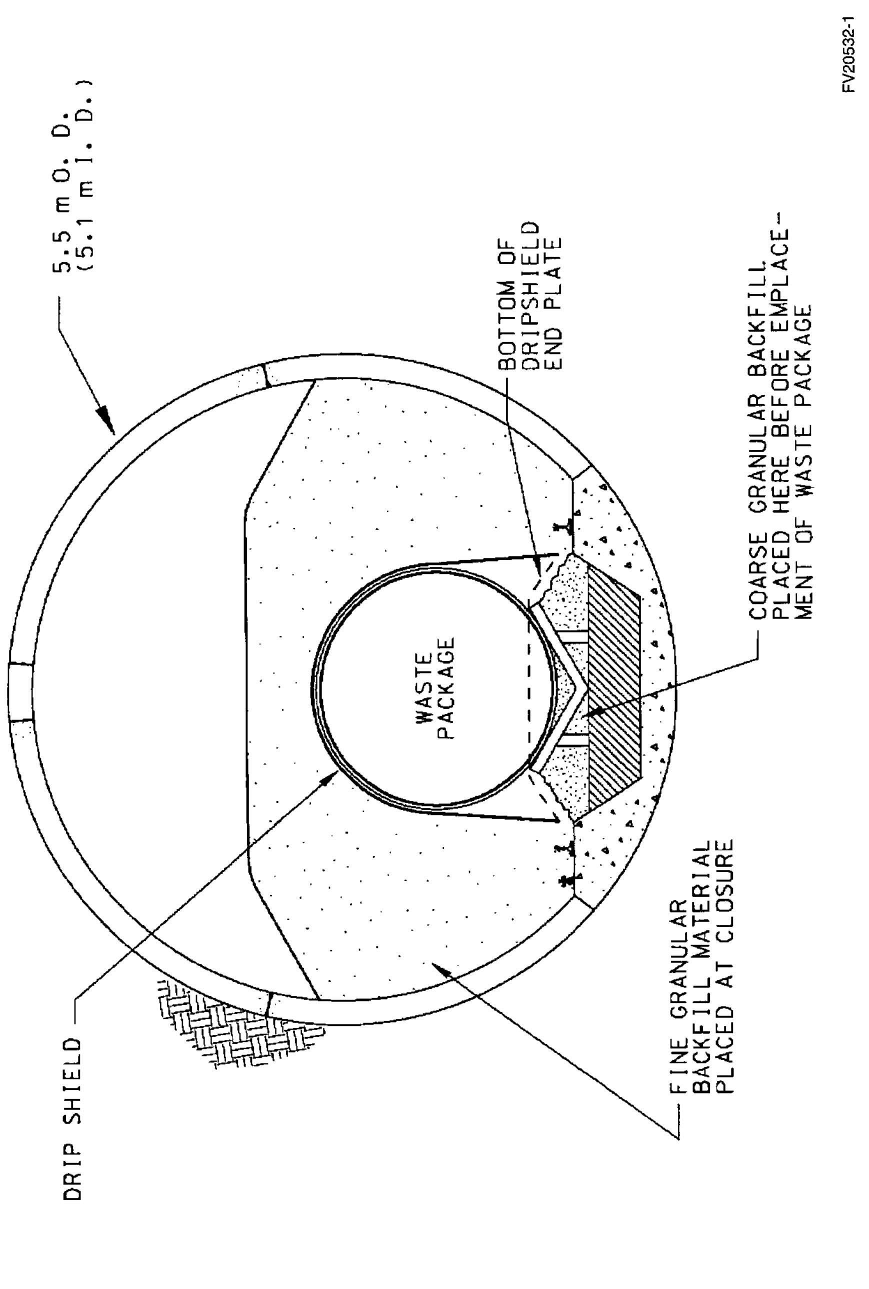


Figure 5-20. Reference Design with Drip Shield and Backfill

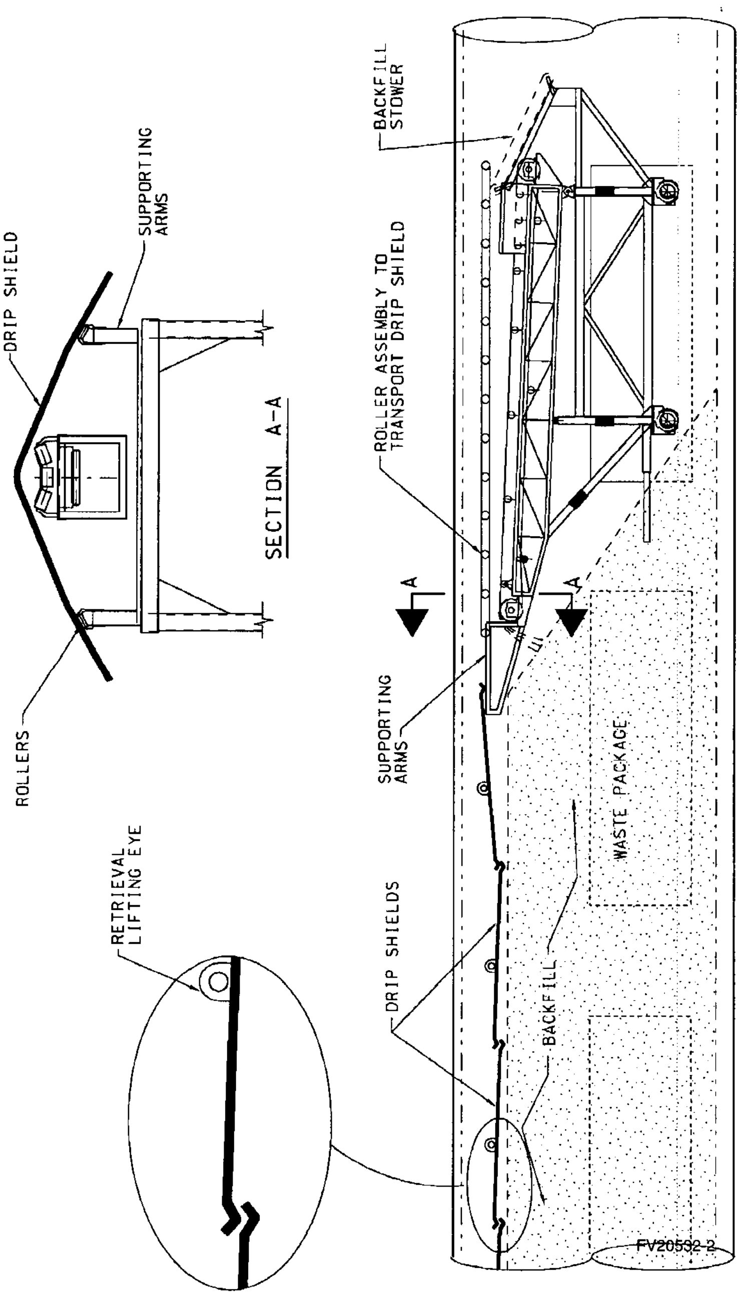


Figure 5-21. Drip Shield Installation

Each of these materials is chemically stable and found in nature. The rates at which they dissolve in water are estimated to be about a few millimeters per million years. Therefore, coatings made of these ceramic materials would likely survive for a million years or more under expected repository conditions.

The major limitation with these materials, however, is the difficulty in forming and handling large, dense shapes. All current ceramic materials are brittle compared with metals. Therefore, a better use of ceramics is as a protective coating. The waste package would benefit from a ceramic coating because the dual-metal construction already makes it strong and resistant to corrosion, and the outer layer of ceramic would add another mechanism to extend the life of the package. The coatings would be applied using a thermal spray technique that uses a ceramic material as the starting or feed material. The type of ceramic feed material and the resulting coating and its properties will vary with the process selected. All of the processes being considered use a high-velocity gas stream plus gas combustion or conversion of electrical energy to melt or partially melt the feed material and spray it onto a surface. When the molten droplets hit the much larger and cooler material to which they are being applied, they conform to the shape of the material, then cool, solidify, and adhere.

The ceramic feed materials can be melted using various sources of heat: an electric arc; a high-velocity, oxygen-fuel flame; or a confined explosion (detonation gun). To date, several versions of application processes have been investigated, applying different ceramic materials, individually and in combinations. In the tests completed so far, the oxygen-fuel-fired coatings have been the densest and most uniform. However, with some development, the other systems may be able to produce similarly good results.

In many respects, thermal spraying resembles spraying paint, and it is often used in routine applications. For example, the rollers on large printing presses are often coated with ceramic materials to improve their resistance to wear. The delivery portion (gun) of a thermal spray system is approximately the size and weight of a hand-held paint sprayer. Though these guns are attached to much larger power and gas supplies, they can be maneuvered easily, allowing large surfaces and contours to be covered smoothly. Further, as with painting, the thermal spray coating is built up in layers for complete coverage. The difference comes in the number of layers applied. In the case of thermal spray applications, hundreds or thousands of layers of oxide particles can be built up to form a complete ceramic coating. This layering process tends to keep most flaws in the coatings very small and produces a highly uniform and fine-grained covering.

The presence of the coating will not change the way the metal underneath corrodes. However, the rate of corrosion will be slower because the ceramic coating will allow less oxygen to reach the metal. Further, if a substantial portion of a coating were to just disappear, the situation would be no worse than it would have been without a coating. A mathematical model has been developed to show how a porous, thermal sprayed ceramic protects steel against corrosion. When the amount of oxygen available is limited, the corrosion rate will decrease. The model accounts for this reduction in the amount of oxygen available.

The model calculation is highly sensitive to the values assigned to the pore parameters, especially the cylinder diameters. Although a few simple estimates have been made based on microscopic examination, the major uncertainties in applying the model lie in not yet having sufficient data on pore geometry. As those measurements improve, the accuracy of the model predictions will improve.

5.3.3.2 Experiments

The following three analytical methods are being used to assess the ceramic coatings before and after they have been exposed to corrosive conditions:

- Microscopic examination (metallography) to evaluate coating geometry, total porosity, and corrosion behavior

- Electrical-conductivity testing to measure how fast oxygen is transported through the coating to the metal underneath

- Mechanical testing (measurements of bond strength and the effects of impacts) to bound the potential effects of handling the coated waste packages or backfilling the drifts after the waste packages have been emplaced

Specimens of steel coated with ceramic are being tested under various environmental conditions in the long-term corrosion facility at Lawrence Livermore National Laboratory. Sets of identical ceramic-coated specimens are exposed to various corrosive conditions and are removed at intervals to measure the effects on the coatings and underlying metal with time. The test setup and conditions have been described in Section 5.1.4. Some of the coatings have been deliberately slit before testing to demonstrate how corrosion might propagate underneath a damaged coating. To simulate various possible repository environments, there are plans to expose additional samples to humid air and dripping water conditions.

Samples are withdrawn from their corrosive environments at intervals for examination, which involves slicing them open to look at the bond area between the ceramic and metal. Before the samples are opened, any slotted regions are filled with epoxy to trap corrosion products that might be present. The samples are then sectioned across the slot and polished. The metal substrates are treated with acid to reveal the grain structure and then examined microscopically to determine the quantities and locations of corrosion products.

As might be expected, coatings applied using a plasma-spray technique are very porous (about 19 percent porosity) and do not provide complete protection against corrosion. In some cases these coatings even spalled near the slot. However, the much denser coatings on high-velocity oxygen-fuel (approximately 2 percent porosity) and detonation sprayed (approximately 6 percent porosity) samples seem to completely protect the substrate, except where the coating was deliberately damaged. The samples showed no apparent undercutting of any of the dense coatings by corrosion near the slots, suggesting that they have little interconnected porosity and are firmly bonded, and that penetration of oxygen beneath a properly applied, dense coating is very slow.

Like corrosion, electrical conductivity in liquid-filled channels or pores is controlled by how easily dissolved ions (like oxygen) can move through the channels. Wide, straight channels offer less resistance to the motion of ions. This resistance is called impedance. On the other hand, narrow, tortuous channels offer much greater impedance. This direct correlation translates to measuring electrical impedance to give a direct indication of how much corrosion can be improved by a given coating in a particular environment. As an example, if the impedance were 100 times greater through a particular coating than without it, corrosion would be slowed to 1/100 the rate because it would take 100 times as long for the same number of ions to reach the metal.

Electrical conductivity is measured using a potentiostat, a device that can apply precisely controlled voltages, currents, and frequencies of electricity. Measurements have been made on steel samples with no coating, oxygen-fueled coatings (approximately 2 percent porosity), detonation coatings (approximately 6 percent porosity), and conventional plasma-sprayed coatings (approximately 19 percent porosity), all immersed in simulated concentrated J-13 water.

The impedance of the highly porous electric arc coating was not significantly different from that of the bare metal. At both low and high electrical frequencies, the two denser coating types showed impedances that were significantly higher than those from the electric arc applied material, thus providing more protection against corrosion.

Coated specimens are undergoing the following two general types of mechanical testing: bond-strength measurements and impact tests. Measuring bond strength quantifies the adhesive and cohesive properties of the coating. Results to date suggest that the coatings are well bonded and tend to fail within the coating rather than at the interface of the ceramic and the metal substrate. Impact tests support this observation and indicate that damage tends to be localized to the area of impact. Impacts with pointed objects are most likely to damage the coating, although any impact severe enough to permanently deform the waste package will cause spalling and can expose the metal. Impact by small stones and gravel should not produce any consequential damage.

5.3.3.3 Design Constraints

Because the ceramic coatings are relatively thin and brittle, a potential exists for the coatings to be damaged as the waste packages are filled with waste and transported and emplaced in the drifts.

The consequences of damaging the protective coating suggest that precautions should be taken to protect the coatings whenever possible. For example, temporary overcoats of polymeric materials or disposable metal sleeves could be applied. Further, using gravel backfill to cover the waste packages would virtually eliminate falling rocks puncturing the coating. Backfill is included as an optional design feature associated with ceramic coatings (see Section 5.3.1).

In addition to the experimental program at Lawrence Livermore National Laboratory, design analyses have been done to determine if ceramic-coated waste packages can meet the structural and thermal requirements necessary for the repository. The structural studies included analyzing the effects of rocks dropping onto a coated waste package, determining if the thermal spray process causes any residual stress that would weaken the package, and evaluating the stresses induced in the coating by the decay heat of the waste. The results of these analyses indicate that the coating will not fracture or spall during normal handling or as a result of the increased temperatures.

In the work done to date, the magnesium aluminate spinel material and high-velocity oxygen application process have produced the most promising coatings. The detonation coatings are nearly as dense, although there is evidence that this spray process leaves an undesirable residue of metal particles. Tests also show that the electric arc techniques can be controlled to produce coatings of higher density (less than the 19 percent porosity thus far demonstrated).

Assuming similar coating performance, an economic and logistical analysis will be required to determine which thermal spray process will best lend itself to the needs of the YMP. The immediate future goals are to continue the evaluations currently in progress. More extensive impedance measurements will be done, including testing materials that have been sealed using various inorganic and metallic sealant materials. If the corrosion work continues to be as promising as early results indicate, there will be a shift toward using larger samples to demonstrate that high density coatings can be applied to larger surfaces, such as those of the waste package designs. An attempt will be made to apply nondestructive evaluation techniques to larger coated surfaces and determine the size and severity of flaws that may be buried in the coatings. Various methods will be used to artificially implant flaws of various sizes in known locations.

6. CONCEPTS FOR CONSTRUCTION, OPERATION, MONITORING, AND CLOSURE

The proposed repository as depicted in the VA would be constructed, developed, and operated in phases, as discussed in the *Mined Geologic Disposal System Concept of Operations* (CRWMS M&O 1997o, Section 2.1), see Section 2 of Volume 5. The Exploratory Studies Facility and surface-based boreholes, trenches, and test pits were constructed during the site characterization phase. The primary focus of this section is on construction and operations.

The construction phase includes the following:

- Transitioning portions of the Exploratory Studies Facility to repository facilities (see Section 4.2.1)
- Constructing surface facilities (see Section 4.1)
- Refurbishing Exploratory Studies Facility openings (see Section 4.2.1)
- Continuing to excavate and equip subsurface facilities (see Section 4.2.1.4)
- Gathering data to support predictions of the repository performance
- Demonstrating some repository operations
- Updating the LA to seek a license to receive, package, and emplace waste

Construction will start about 5 years before waste receipt and emplacement begins. Construction of the underground facility will continue during the waste handling and emplacement operations phase. The expected duration of this concurrent development and emplacement phase is approximately 20 years.

The waste handling and emplacement operations phase includes the following:

- Receiving the waste
- Packaging the waste in disposal containers
- Emplacing loaded, sealed, and tested disposal containers (waste packages) in the repository
- Performing initial monitoring and maintenance activities
- Gathering data to support predictions of repository performance

The duration of the waste handling and emplacement operations phase is estimated at 24 years.

6.1 MONITORED GEOLOGIC REPOSITORY CONSTRUCTION

The construction of initial portions of the surface and subsurface facilities are described below. These operations include additions to the existing surface facilities, retrofitting the north and south portals, north and south ramps, and the east main drift, muck handling during excavation, and the installation of the subsurface ventilation systems. The simultaneous operations of ongoing drift excavation and waste emplacement are also described.

6.1.1 Surface Construction

After NRC authorizes construction, surface construction will begin with site preparation, which includes grading and grubbing; establishing building pads, rails, roads, and drainage; and constructing water, security, support, and laydown facilities. As many of the existing facilities as possible will be preserved and used during these initial activities. The facilities include the changehouse, switchgear building, and substation. In addition, the north portal pad will be rebuilt and enlarged to meet repository design requirements. Following site preparation, facility structures will be built and transportation and security systems will be completed.

6.1.2 Subsurface Construction

The north and south portals and ramps and the east main drift were excavated during construction of the Exploratory Studies Facility. These openings will be retrofitted, as necessary, during repository construction. Cross-block access drifts and perimeter main drifts will be excavated to permit excavation of the emplacement drifts. The underground service drifts, ventilation shafts, and approximately 5 percent of the emplacement drifts will be completed before repository operations begin (see Section 4.2.1.4). Figure 6-1 shows the configuration of the facility when waste emplacement begins.

Initially, subsurface excavation will be supported from the north and south portals using the refurbished site characterization facilities. Before waste emplacement begins, development operations will shift to the south portal exclusively.

Following initial construction, underground openings will be developed concurrently with waste emplacement operations. The development will not interfere with waste emplacement operations. Isolation air locks will maintain physical and functional separation of the two activities. When a predetermined number of newly excavated emplacement drifts are ready for waste emplacement, the isolation air locks will be moved to include the newly developed drifts. The required development rate for emplacement drifts will be affected by several factors including the number of emplacement drifts, emplacement strategy, waste package spacing, emplacement drift spacing, and the number of tunnel boring machines used.

The repository openings are designed to serve a variety of functions. Main accesses (shafts and ramps) provide facilities for ventilating the subsurface, emplacing waste, removing excavated material, performing maintenance, and transporting personnel and materials. Tunnel boring machines will be used for most underground excavations. Where use of a tunnel boring machine is not feasible, other mechanical methods such as roadheader machines may be used.

After a drift is excavated, worker safety will be maintained by scaling loose rock in combination with appropriate ground supports (rockbolts, welded wire fabric, cast-in-place concrete, and segmented precast concrete linings).

A conveyor belt will transport excavated rock (muck) from the subsurface to the surface. The procedure for loading the muck onto the conveyor belt will depend on the method of excavation. Both the tunnel boring machine and roadheader have internal gathering systems and conveyors for loading and transferring the muck. These systems may be used to transfer muck to rail cars that are transported to a muck dump. Muck transport methods are generally limited by the maximum allowable particle size. Depending on the transport method, rail car or conveyor, some of the muck may require crushing to meet the size limits.

Crushing would most likely be required for conveyor transport, which is the preferred method for ramps and drifts, and least likely to be true for rail transport. Drill-and-blast excavation is the method most likely to produce oversized particles. All excavation methods may require large rock pieces to be scaled from the roof of the opening. At the surface, the excavated rock will be placed in a storage pile.

Aesthetic, environmental, and economic factors will be considered in determining how the muck should be processed and where the storage pile should be located.

During pre-emplacement operations, an emplacement drift and an emplacement drift turnout will be prepared. These activities may include the following:

- Preparing equipment
- Installing drift access control doors
- Preparing utilities
- Installing emplacement and turnout ventilation system

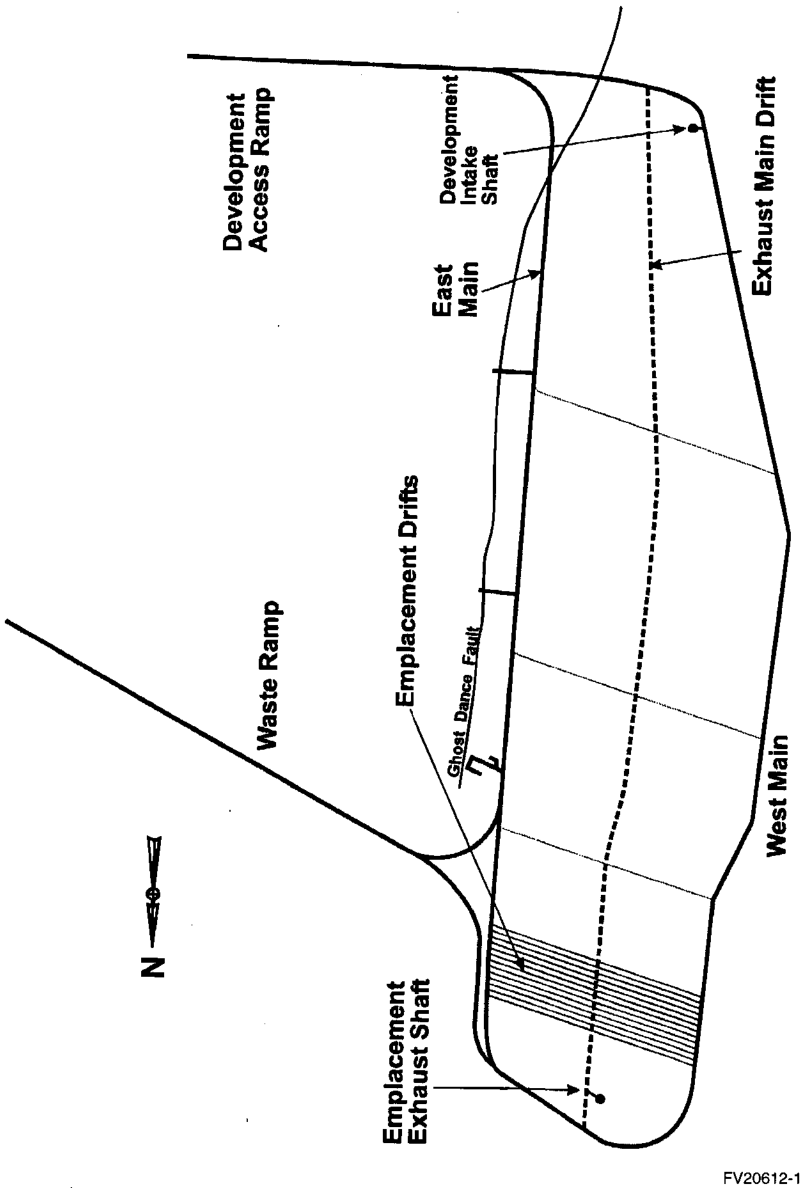


Figure 6-1. Subsurface Layout when Emplacement Begins

- Installing shielding
- Installing control devices and monitoring instrumentation
- Installing inverts, piers, and supports
- Final testing and equipment checkout before beginning emplacement
- Installing the rails for waste transport

6.2 WASTE HANDLING AND EMPLACEMENT OPERATIONS

This section provides an overview of the operations for receiving transportation casks, waste handling and packaging, and placing waste packages in the emplacement drift. Additional details on equipment discussed in this section can be found in Sections 4.1 and 4.2 for surface and subsurface facilities, respectively.

6.2.1 Surface Operations

Transportation casks containing canistered and uncanistered commercial spent nuclear fuel, canistered DOE and naval spent nuclear fuel, or canistered vitrified high-level radioactive waste (see Figure 6-2) will be delivered to the repository by road and/or rail. Each transportation cask is expected to be equipped with impact limiters and personnel barriers.

Arriving transportation cask carriers and their offsite prime movers will be inspected at the repository security gate for contraband, sabotage, and contamination. An onsite prime mover will transfer the loaded cask carrier to the truck or rail parking area within the radiologically controlled area. The offsite prime mover will wait within the radiologically controlled area to pick up an empty cask carrier for subsequent shipment back to a waste generator.

The onsite prime mover will move the loaded cask carriers from a parking area to the Carrier Preparation Building. The cask and carrier will be inspected for radiological surface contamination, the personnel barrier will be retracted or removed, the impact limiters will be removed from the cask, and the cask will be reinspected for radiological surface contamination. Minor decontamination of the cask will be performed as required. Further decontamination of the cask will be performed within the Waste Handling Building, if required.

The prepared cask carrier will either remain in the Carrier Preparation Building or may be staged in a parking area until the Waste Handling Building is available. The transportation cask will be removed from the cask carrier in the Waste Handling Building.

After the transportation cask has been unloaded, the empty cask will be loaded onto the cask carrier, and the cask carrier will be returned to the Carrier Preparation Building. The personnel barriers and impact limiters will be reinstalled to support subsequent transportation back to the waste generator. Incidental cask maintenance will be performed at the Carrier Preparation Building.

In the Waste Handling Building, the transportation cask will be removed from the carrier, the cask will be opened, and the waste (assemblies or disposable canisters) will be placed into disposal containers. Then the disposal containers will be sealed, tested, and prepared for transportation underground.

Transportation casks will be received at the carrier bay in the Waste Handling Building. A bridge crane (Figure 4-8) will lift the cask to an upright position and place it on a cask transfer cart. The cart will be assigned to either one of three assembly transfer lines or one of two canister transfer lines, as appropriate. Casks containing uncanistered spent nuclear fuel or dual-purpose canisters (dual-purpose canisters are not suitable for disposal) will be routed to the pool assembly transfer lines. Casks containing disposable canisters will be routed to the dry canister transfer lines.

The carrier bay will also receive empty transportation casks and empty overpacked dual-purpose canisters from the cask preparation areas and load them onto carriers for reshipment. The empty overpacked dual-purpose canisters may be either

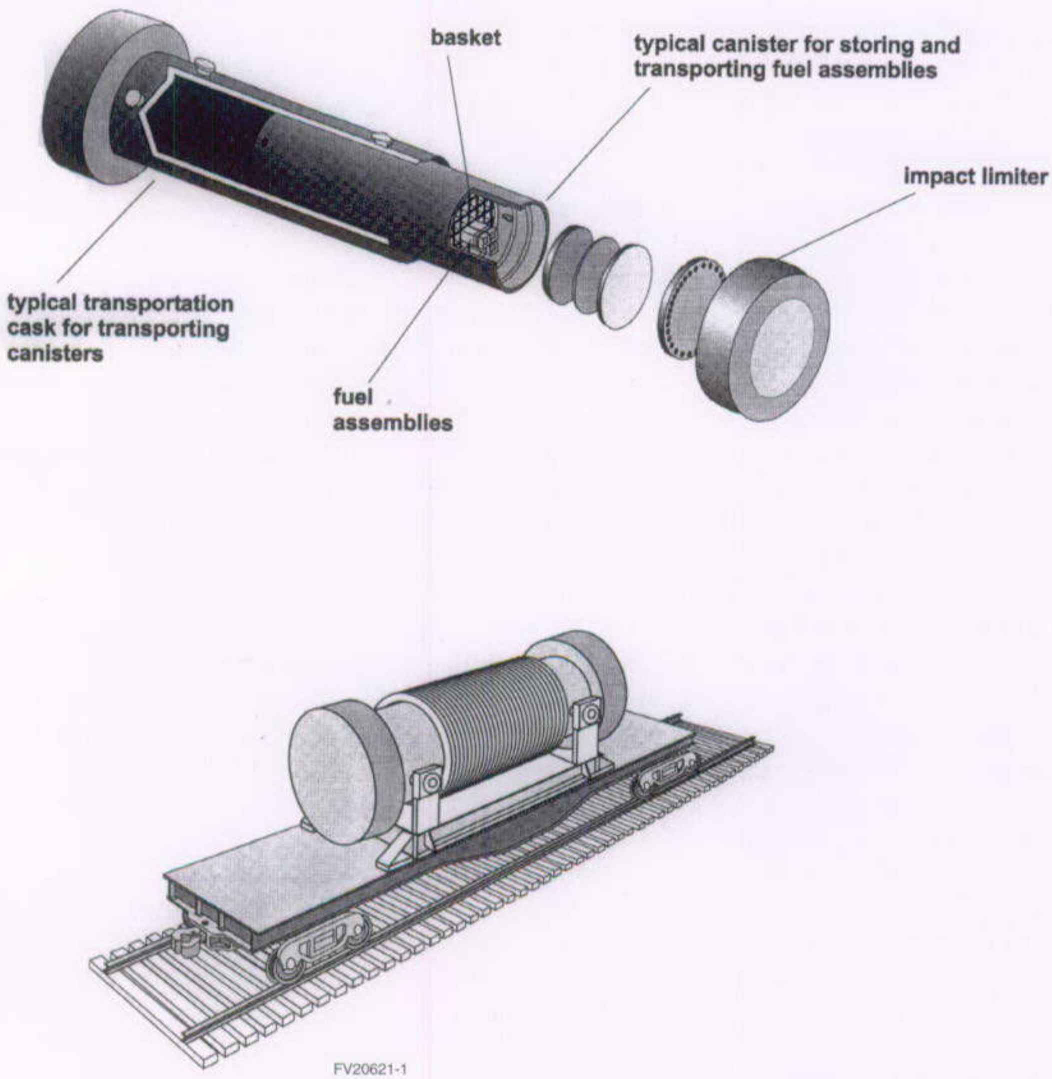


Figure 6-2. Transportation Casks

shipped for disposal as low-level radioactive waste or returned to the supplier if the dual-purpose canister is reusable.

The assembly transfer system (Figures 4-9a and b) will unload spent nuclear fuel assemblies and dual-purpose canisters from the casks and load the assemblies into the disposal containers. For spent nuclear fuel received in dual-purpose canisters, the assemblies will be removed from the canister after it has been removed from the transportation cask. The assembly transfer system will be equipped with three identical assembly transfer lines that are capable of operating concurrently.

The casks will be positioned into a cask preparation pit where they will be cleaned as required, the internal pressure will be measured, and the internal gas will be sampled and analyzed. Casks containing uncanistered fuel assemblies will be prepared for unloading by cooling, filling with water, and removing the lid bolts. A lid lift fixture and cask lift yoke will then be installed. The cask will be transferred and lowered into a cask unload pool and the lid will be removed. The lift fixture and yoke will be returned to the cask preparation area. For casks containing dual-purpose canisters, the cask lid will be removed and the canister will be prepared for unloading by measuring the pressure within the canister, sampling the internal gas, cooling, and filling the canister with water. The dual-purpose canister lift fixture and cask lift yoke will then be attached. The cask containing the dual-purpose canister will be transferred to and lowered into the pool, where the canister will be removed from the cask. The dual-purpose canister will then be placed in an overpack container, a lid severing tool will be installed, and the canister lid will be cut off.

The spent nuclear fuel assemblies will be unloaded using a wet assembly transfer machine. Following unloading, the overpack containing the lower section of the canister will be drained and returned to the preparation area.

Empty casks will be returned to the cask preparation area and prepared for offsite shipment. Lids will be installed on the casks, pool water removed, and the cask dried. The cask will then be decontaminated, surveyed, and returned to the carrier/cask handling system for offsite shipment for reuse. Used dual-purpose canister sections (including the lids) and the overpack will be washed with demineralized water at the pool. The dual-purpose canister and the overpack will then be transferred to the cask preparation pit, where the canister and overpack will be drained and cleaned, and the lid replaced on the dual-purpose canister. Finally, the lid will be installed on the overpack and bolted. The unit will then be decontaminated and surveyed prior to shipment offsite.

The wet assembly transfer machine will transfer unloaded spent nuclear fuel assemblies into baskets staged in the pool. An interconnecting transfer canal will permit the assemblies to be moved among the three pools. When the correct complement of spent nuclear fuel is available for loading into a disposal container, baskets of assemblies will be loaded into inclined basket carts and transported from the pools to one of two assembly drying stations located in a dry assembly transfer cell. A dry assembly transfer machine in the cell will load the baskets into the dryers.

Before being loaded with spent nuclear fuel assemblies, the empty disposal containers will be transferred to the disposal container load cell where the port mating system will be sealed to the top of the empty disposal container to control the spread of contamination. After drying, the assemblies will be transferred one at a time through a loading port and into a disposal container. In addition to inserting the assemblies, it may be necessary to insert control rods into the disposal container to maintain criticality control. The options and concepts for inserting the control rods are being studied. The loaded disposal container will then be transported to a decontamination cell where the lid will be decontaminated and surveyed and the disposal container inerted with nitrogen.

The canister transfer system will open casks containing commercial spent nuclear fuel, defense high-level radioactive waste, and DOE and naval spent nuclear fuel in disposable canisters and transfer the canisters into a disposal container. The transfer system will include two identical canister transfer lines that operate concurrently.

Casks containing disposable canisters will be positioned at a cask preparation cell in a canister transfer line. The casks will be prepared by measuring the internal pressure, analyzing the internal gas, removing the lid bolts, and installing the lid lifting fixture. The cask will then be decontaminated as required and transferred to the lid removal and unloading cell. Once a cask with disposable canister has been prepared and decontaminated, the cask lid will be removed using the cask lid lifting fixture, and a large or small canister lifting fixture will be attached to the canisters inside the cask. The disposable canister will then be transferred into the disposal container and readied for closure.

New disposal containers will be transported by rail to the empty disposal container handling area located in the disposal container receiving shed. The option to receive new disposal containers in the Waste Handling Building is also being analyzed. Disposal container lifting and base collars will be installed and the empty disposal container disengaged from the carrier. A disposal container lift beam assembly will then be installed and the disposal container transferred to a tilting station using a bridge crane. The disposal container will be raised to a vertical orientation, outfitted with a lid and other required furnishings, and staged. Disposal containers scheduled for loading will be moved through the disposal container welding cell to a waste transfer line using the transfer carts and the bridge crane.

Large canisters will be transferred and loaded into an empty disposal container in the disposal container loading area. Small canisters can either be loaded directly into a disposal container or accumulated in a staging rack until enough compatible canisters are available to fill a disposal containers. The loaded disposal container will then be mounted on a disposal container transfer cart and transported to the disposal container handling system.

Loaded disposal containers will enter the disposal container welding stations (Figure 4-11) where they will be mounted on a turntable with a bridge crane. The inner lid will be installed, welded in place, and a welding/inspection robot will conduct a nondestructive examination of the first seal. The disposal container will then be inerted. Then the outer disposal container lid will be installed, welded, and a second nondestructive examination conducted for the second seal. The bridge crane will transfer the loaded waste package either to the staging area or to the tilting station. At the tilting station, the waste package will be lowered onto a horizontal transfer cart, and the lifting and base collars will be removed. Then the waste package will be moved to the waste package transfer/decontamination area, lifted by a horizontal lifting system, decontaminated, and lowered onto a reusable rail car for insertion into the waste package transporter.

Off-normal operations are defined as operations, other than normal operations, that take place during Category 1 and Category 2 design basis events. Category 1 design basis events are natural- and human-induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before the repository operations area is permanently closed. Category 2 design basis events are other natural- and human-induced events that are considered unlikely, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety.

Following are examples of surface design basis events:

- A primary handling or confinement component failure to perform its intended function on demand; loss of utility power for a limited duration

- Shipping casks, disposable or nondisposable canisters, or waste packages that are out of specification or cannot be handled without corrective measures, such as decontamination, replacing a seal, or tightening bolts

- Spent nuclear fuel assembly handling events